

much anyway. You would just as soon leave the contaminated surface and add a new one over it. On the other hand, we have not considered the possibilities of trying to decontaminate the surface. It is rather an involved problem. It is like trying to decontaminate a filter. Usually we bury them, or burn them, and try to reduce their volume that way.

If the board has a significant deposition of fission product particulates, I would be afraid of the problem of decontamination. It would be pretty serious.

Panel Chairman: If there are no others, I would first remind you that tomorrow Session V starts at 8:30 A.M., and, secondly, I would like to thank our four speakers for an excellent group of papers.

## SESSION V - CONTAINMENT AND CONFINEMENT APPROACHES

Morning - 23 October 1963

M. D. Thaxter, LRL, Chairman

Session Chairman: Good morning, gentlemen. This is Session V, Containment and Confinement Approaches. We have been scheduled to start at 8:30 A.M.

The first paper this morning is by Mr. T. D. Anderson of ORNL with the title, "The Holdup Effect of Double Reactor-Containment and Its Influence on Dose from Airborne Radioactive Materials."

# THE HOLDUP EFFECT OF DOUBLE REACTOR-CONTAINMENT AND ITS INFLUENCE ON DOSE FROM AIRBORNE RADIOACTIVE MATERIALS

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## Introduction

Following a reactor accident the safety problem of greatest concern is the release of airborne radioactive materials to the environment. A design feature that would reduce such activity release is a second containment structure that surrounds the primary containment vessel. Specific examples of plants that utilize double containment are the Indian Point and the N.S. SAVANNAH Reactors. In these plants the space between the first and second containers is maintained at sub-atmospheric pressure by an exhaust blower connected in series with an air cleaning system. Thus, all material that leaks into the secondary containment space must pass through the air cleaning system before reaching the atmosphere.

Controlled activity release through an air cleaning system would result in a significant reduction in the release of those materials for which the air cleaning system is effective. An additional effect, which is the one of present interest, would be the time delay in release introduced by the second container. This time delay would result in a short-term reduction in the dose from all airborne radioactive materials; and, if decay of the isotope causing the dose were sufficiently rapid, a permanent reduction in dose would result. Even a temporary reduction in activity release following a reactor accident could be significant since the additional time available could be used for measures that would reduce the consequences of the accident. The double containment holdup effect could be especially important for ship reactors since it would be possible to move the system away from populated areas shortly after the accident.

The extent to which the holdup effect would alleviate environmental hazards resulting from a reactor accident would depend on the containment system parameters, the characteristics of the isotope causing the dose, and the manner in which radioactive materials were released from the reactor. The purpose of the present investigation was to evaluate the effectiveness of double-containment holdup in reducing environmental hazards associated with the release of radioactive materials after a reactor accident.

## Containment Model

The containment model used in the present analysis is shown in Fig. 1. Basically the model is made up of three series-connected enclosures consisting of the reactor core and two containment structures. As may be seen from Fig. 1, an isotope can enter an enclosure by leakage from the preceding enclosure and by decay of the precursor isotope. Material can leave an enclosure by leakage and by decay to a daughter isotope. Deposition effects were not included in the containment model.

The leakage from a containment structure is normally characterized by a leak-rate constant which gives the fraction of the contained material that leaks out in unit time. The implication of this practice is that the rate of leakage of a given gaseous material is proportional to the total quantity of the material enclosed by the containment structure. This assumption forms the basis of the analysis given here.

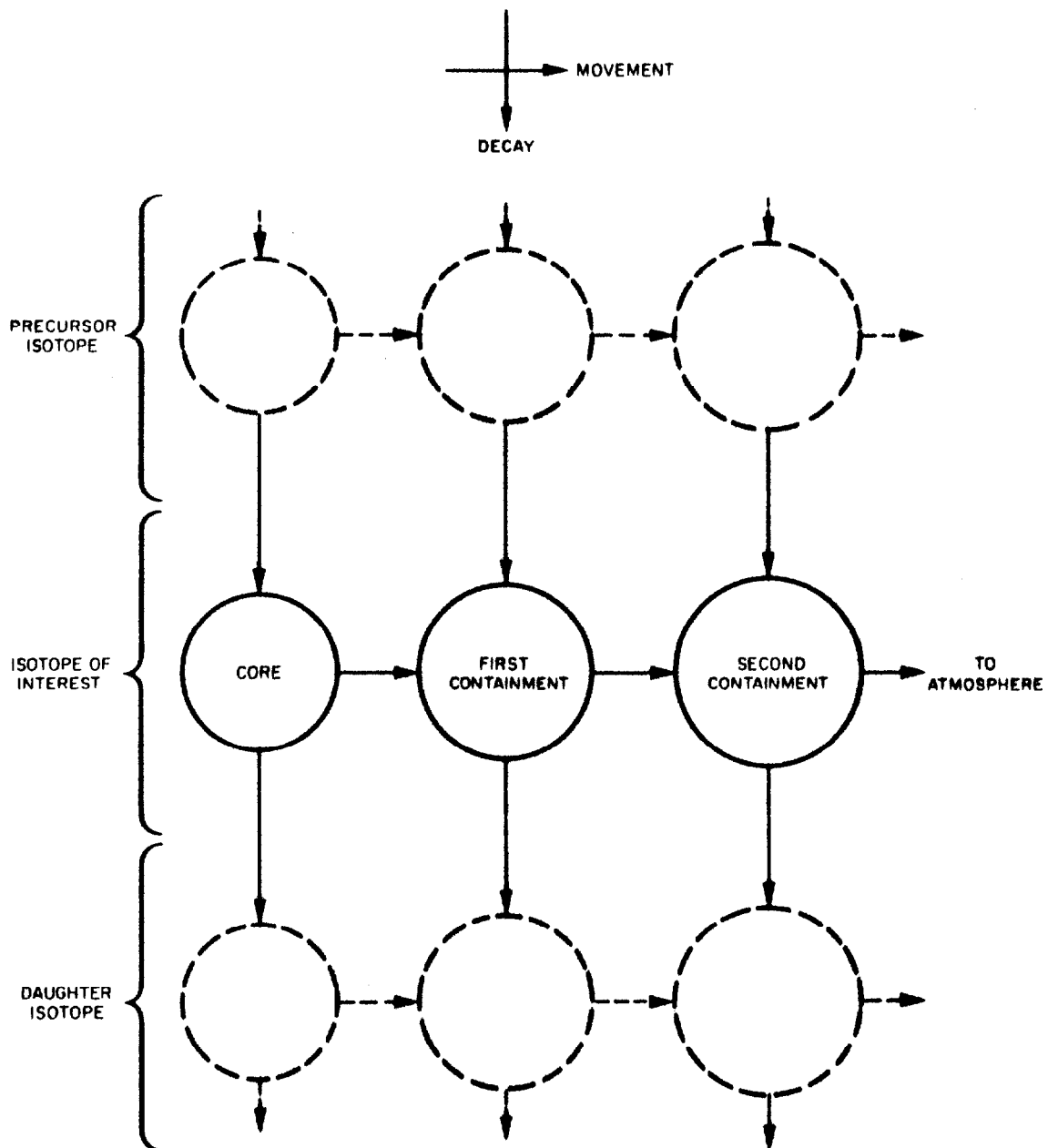


Fig. 1. Schematic of Containment System.

The containment rate processes for a given isotope are shown in Fig. 2. The symbols used in Fig. 2 are defined as follows:

- $F(t)$  = fraction of the total quantity of the isotope that is outside the reactor core at time  $t$ ,  
 $l$  = leak-rate constant for first containment structure,  
 $m$  = leak-rate constant for second containment structure,  
 $N(t)$  = total quantity of the isotope in existence at time  $t$ ,  
 $Q(t)$  = quantity of the isotope held up in the first containment structure,  
 $R(t)$  = rate of release of the isotope from the reactor core,\*  
 $U(t)$  = quantity of the isotope held up in the second containment structure, and  
 $\lambda$  = decay constant of isotope.

The subscript,  $p$ , is used to denote the precursor isotope.

### Leak-Rate Equations

A macroscopic material balance gives the following differential equations for the quantity of material held up in the containment structures:

$$\frac{dQ(t)}{dt} + (l + \lambda) Q(t) = N(t) \frac{dF(t)}{dt} + [F(t) - F_p(t)] \lambda_p N_p(t) + \lambda_p Q_p(t) \quad (1)$$

and

$$\frac{dU(t)}{dt} + (m + \lambda) U(t) = lQ(t) + \lambda_p U_p(t) \quad (2)$$

As is evident from Eqs. (1) and (2) the amount of an isotope in a given container depends on the amount of the precursor isotope in that container. There is, of course, a set of equations similar to Eqs. (1) and (2) for each precursor isotope. Thus, in general, the solutions to (1) and (2) can be obtained only after the solutions for the precursors are obtained. It can be shown,<sup>1</sup> however, that the solutions to (1) and (2) are considerably simplified if all precursor isotopes move through the system in the same manner as the isotope of interest. With this condition the problems of mobility and decay are separable, and the solutions to (1) and (2) are

$$Q(t) = \left[ F(t) - l e^{-lt} \int_0^t e^{l\tau} F(\tau) d\tau \right] N(t) \quad (3)$$

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\*The rate of release,  $R(t)$ , is related to the fraction released,  $F(t)$ , by<sup>1</sup>

$$R(t) = N(t) \frac{dF(t)}{dt} + [F(t) - F_p(t)] \lambda_p N_p(t) .$$

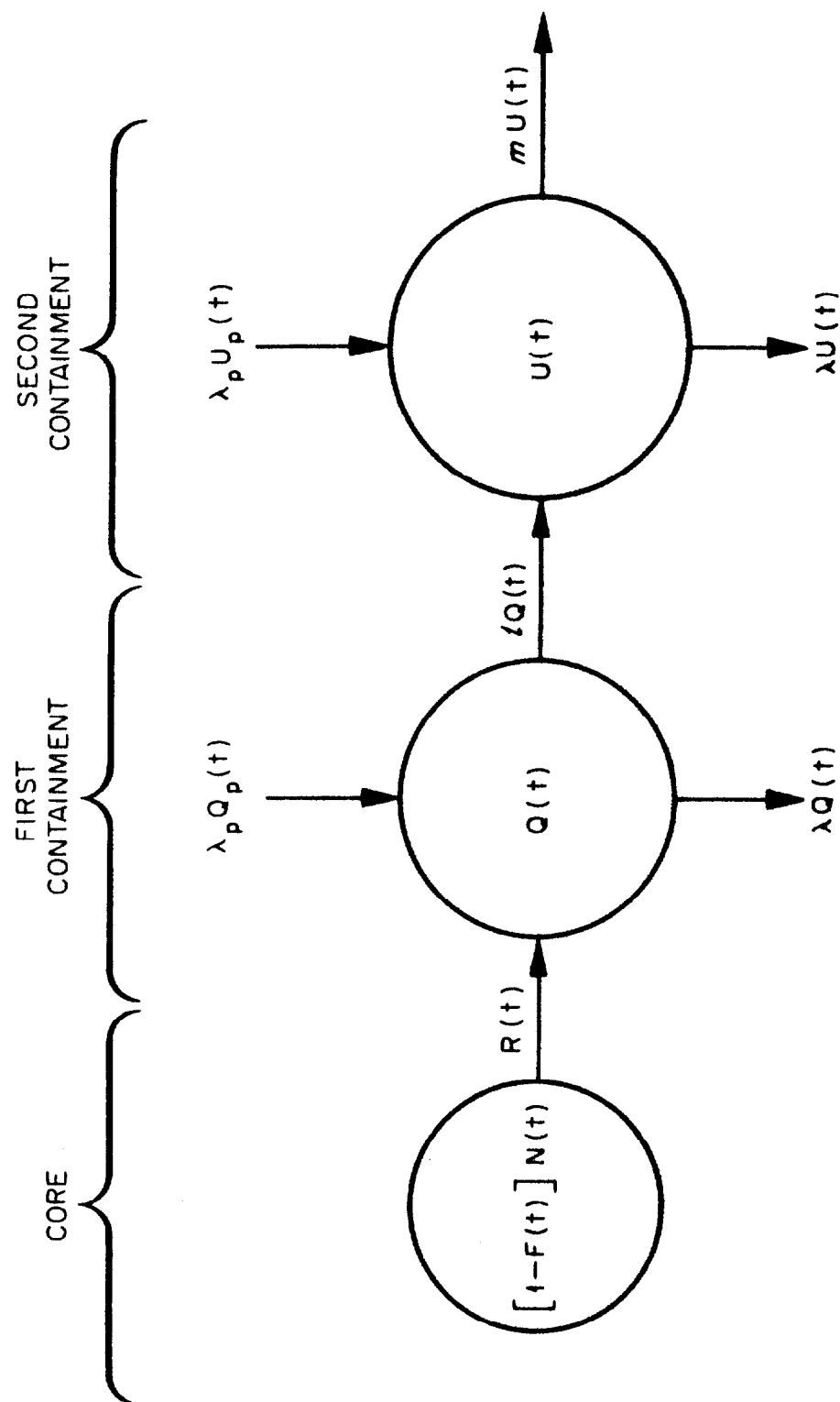


Fig. 2. Containment Rate Processes.

and

$$U(t) = \frac{l}{m-l} \left[ m e^{-mt} \int_0^t e^{m\tau} F(\tau) d\tau - l e^{-lt} \int_0^t e^{l\tau} F(\tau) d\tau \right] N(t) . \quad (4)$$

The rate of leakage from the first container,  $S_1(t)$ , is given by

$$S_1(t) = l Q(t) = l \left[ F(t) - l e^{-lt} \int_0^t e^{l\tau} F(\tau) d\tau \right] N(t) . \quad (5)$$

The leak rate from the second container,  $S(t)$ , is given by

$$S(t) = m U(t) = \frac{ml}{m-l} \left[ m e^{-mt} \int_0^t e^{m\tau} F(\tau) d\tau - l e^{-lt} \int_0^t e^{l\tau} F(\tau) d\tau \right] N(t) \quad (6)$$

#### Leak-Rate with Step Release from Core

To evaluate the leak rates given by Eqs. (5) and (6) it is necessary to know the fraction of the isotope released from the core,  $F(t)$ . In safety analyses of power reactors it is normally assumed that some fraction,  $f$ , of the total quantity of a given fission product is released instantaneously from the core at time  $t_0$  after the start of the accident. With this core release model,  $F(t)$  is given by

$$F(t) = \begin{cases} 0 & t \leq t_0 \\ f & t > t_0 \end{cases} ,$$

and Eqs. (5) and (6) become

$$S_1(t) = l f e^{-l(t-t_0)} N(t), \quad t > t_0 , \quad (7)$$

and

$$S(t) = l f \frac{m/l}{m/l - 1} \left[ e^{-l(t-t_0)} - e^{-m(t-t_0)} \right] N(t) , \quad t > t_0 . \quad (8)$$

For convenience in graphical presentation the leak rate to the environment given by Eq. (8) can be arranged in the dimensionless form

$$\frac{S(t)}{l f N(t)} = L(\theta) ,$$

where

$$L(\theta) = \frac{\kappa}{\kappa-1} \left[ e^{-\theta} - e^{-\kappa\theta} \right] ,$$

$$\theta = l(t - t_0) ,$$

and

$$\kappa = \frac{m}{l} .$$

The relative leak rate,  $L(\theta)$ , is shown graphically in Fig. 3 for a range of  $\theta$  and for several values of the parameter,  $\kappa$ .

#### Connection Between Release-Rate and Dose

The dose rate from airborne radioactive material is proportional to the air concentration of the material at or near the receptor point. The air concentration is, according to atmospheric dispersion formulae, proportional to the release rate with the condition that negligible time elapses between release and arrival at the receptor point. The dose rate is, therefore, proportional to the release rate. For a given isotope the dose rate,  $\dot{D}(t)$ , is

$$\dot{D}(t) = K S(t)$$

where  $S(t)$  is the release rate to the atmosphere and  $K$  is a time independent quantity determined by the isotope under consideration, the atmospheric conditions, and the position of the receptor point relative to the release point. The dose,  $D(t)$ , up to time,  $t$ , is given by

$$D(t) = \int_0^t \dot{D}(\tau) d\tau = K \int_0^t S(\tau) d\tau . \quad (10)$$

Using Eq. (7) in Eq. (10) it is found that the dose,  $D_1(t)$ , from a single containment system with step release of material from the core is

$$D_1(t) = lfK \int_{t_0}^t e^{-l(\tau - t_0)} N(\tau) d\tau . \quad (11)$$

Similarly using Eq. (8) in Eq. (10) the double containment dose is seen to be

$$D(t) = lfK \frac{m/l}{m/l - 1} \int_{t_0}^t \left[ e^{-l(\tau - t_0)} - e^{-m(\tau - t_0)} \right] N(\tau) d\tau . \quad (12)$$

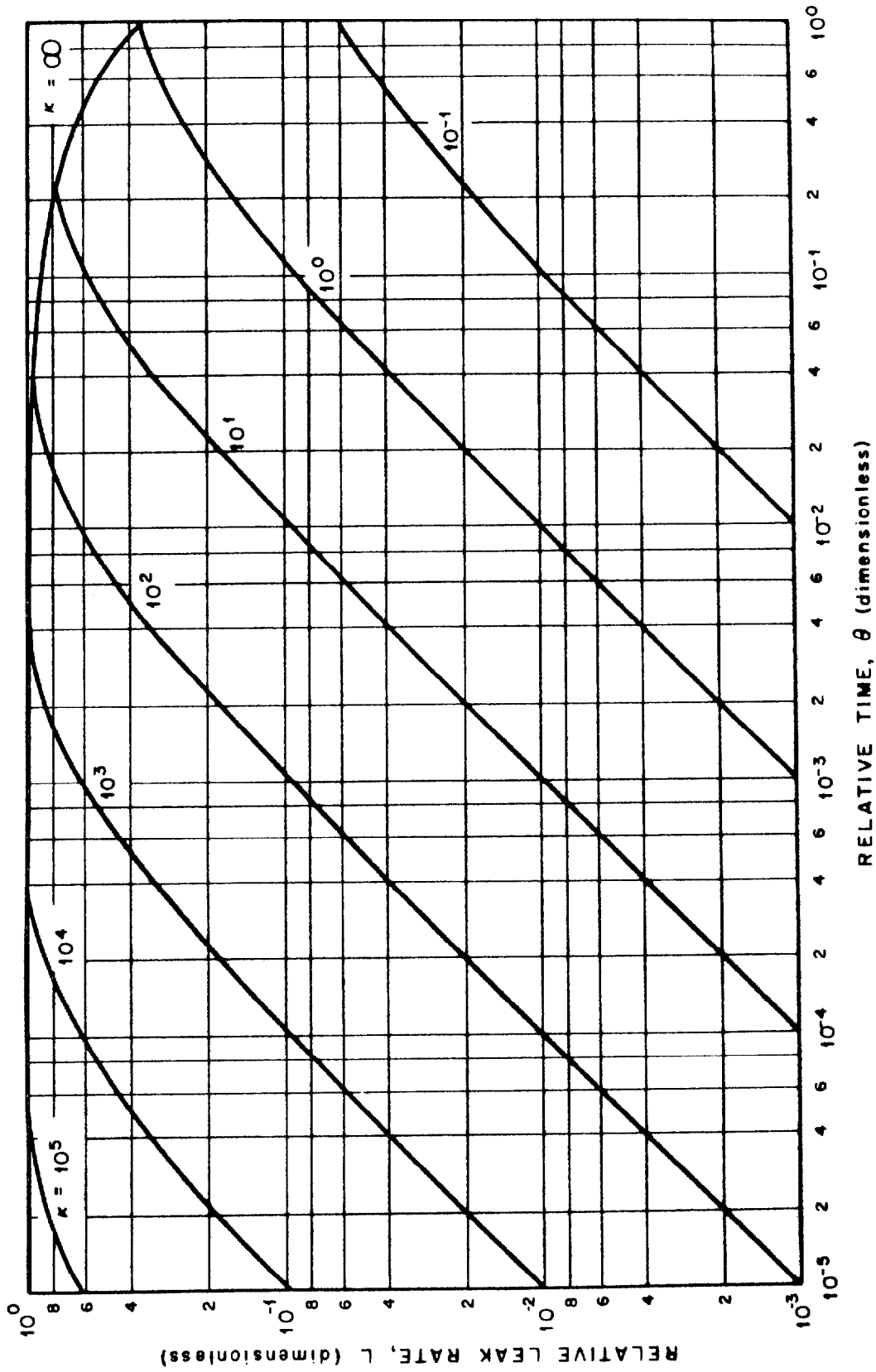


Fig. 3. Leak Rate from Double-Containment System With Step Release from Core.

The use of two series-connected containment structures can effect a reduction in the exposure dose arising from a reactor accident. The extent of this reduction can be determined by taking the ratio of the dose with double containment to the dose with single containment. Using the step model of core release, the dose ratio given by dividing Eq. (12) by Eq. (11) is

$$\frac{D(t)}{D_1(t)} = \frac{m/l}{m/l - 1} \left[ 1 - \frac{\int_{t_0}^t e^{-m(\tau - t_0)} N(\tau) d\tau}{\int_{t_0}^t e^{-l(\tau - t_0)} N(\tau) d\tau} \right]. \quad (13)$$

From (13) it is seen that the reduction in dose depends on  $N(t)$  as well as the leak characteristics of the containment structures. Because of precursor isotopes, the time dependence of  $N(t)$  depends on the neutron flux level and the operating time of the reactor prior to the accident. The general character of the effect of double containment can be demonstrated, however, without introducing the reactor details. This is done by assuming simple exponential decay of the isotope causing the dose. For this case

$$N(t) = N_0 e^{-\lambda t}$$

and Eq. (13) becomes

$$\frac{D(t)}{D_1(t)} = \frac{\kappa}{\kappa - 1} \left[ 1 - \frac{(1 + \kappa \alpha)}{(1 + \alpha)\kappa} \cdot \frac{1 - e^{-(1 + \alpha)\kappa \theta}}{1 - e^{-(1 + \kappa \alpha)U}} \right] \quad (14)$$

where  $\alpha = \frac{\lambda}{m}$  and the other quantities are as defined previously. For materials that undergo radioactive decay it would be expected that the holdup effect of the second containment would result in a permanent reduction in the exposure dose. The infinite-time dose ratio from Eq. (14) is

$$\left[ \frac{D(t)}{D_1(t)} \right]_{t=\infty} = \frac{1}{1 + \alpha} \quad (15)$$

Thus, the permanent reduction in dose because of double containment is dependent only on the ratio of the decay constant to the leak-rate constant for the second container. The infinite-time dose ratio given by (15) is shown graphically in Fig. 4. It can be seen that when the decay constant,  $\lambda$ , is small relative to the leak-rate constant,  $m$ , there is no permanent reduction in the dose. There is, however, a temporary reduction in dose even if there is negligible decay ( $\lambda \sim 0$ ) of the isotope producing the dose. To illustrate this effect the time-dependent dose ratio given by Eq. (14) is shown in Fig. 5 for  $\alpha=0$  (no decay,  $\lambda=0$ ), for  $\alpha=1$  ( $\lambda=m$ ), and for two values of  $\kappa$ . It can be seen that the short-time effectiveness of double containment in reducing the dose is little affected

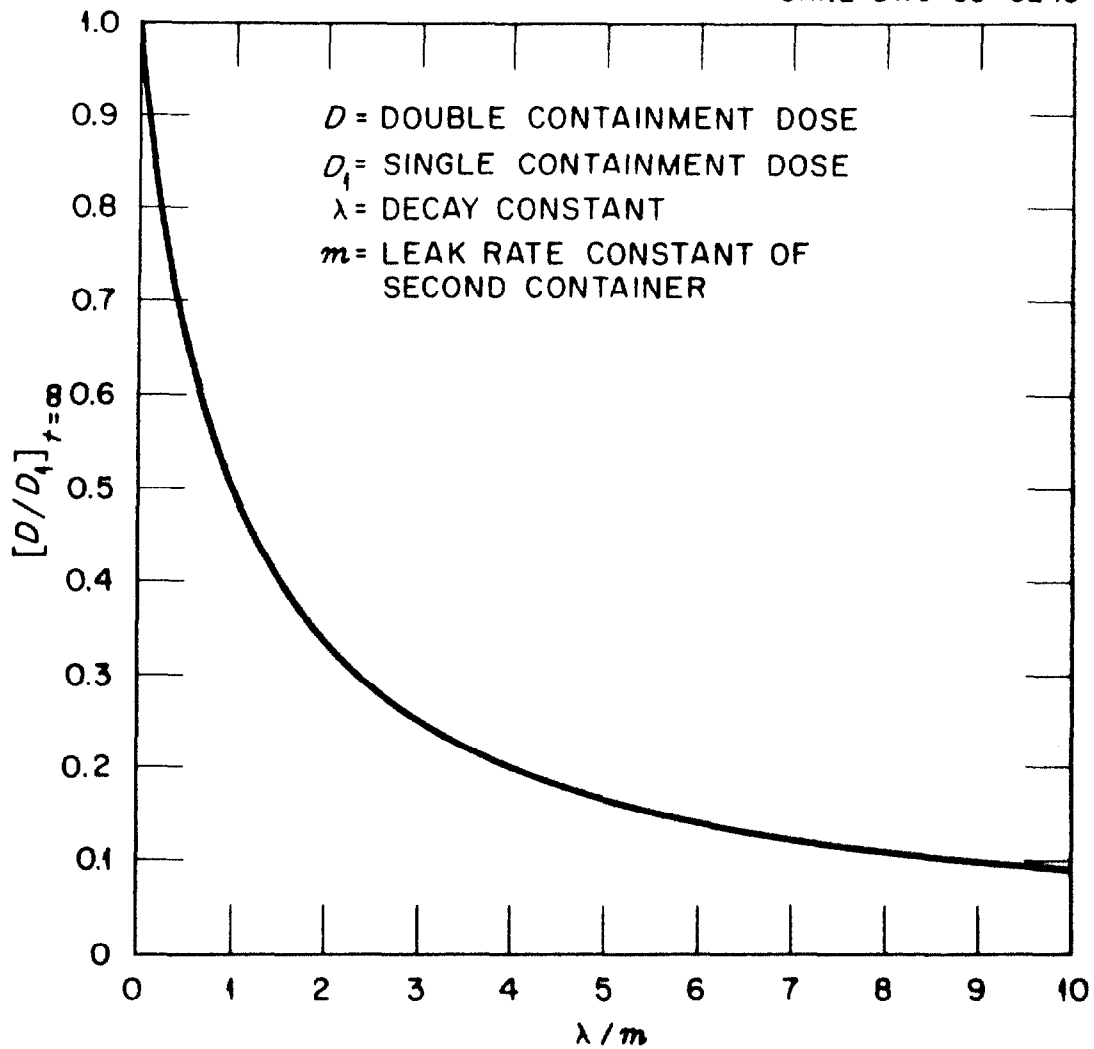


Fig. 4. Effect of Double Containment on Infinite-Time Dose from Isotope Decaying Exponentially.

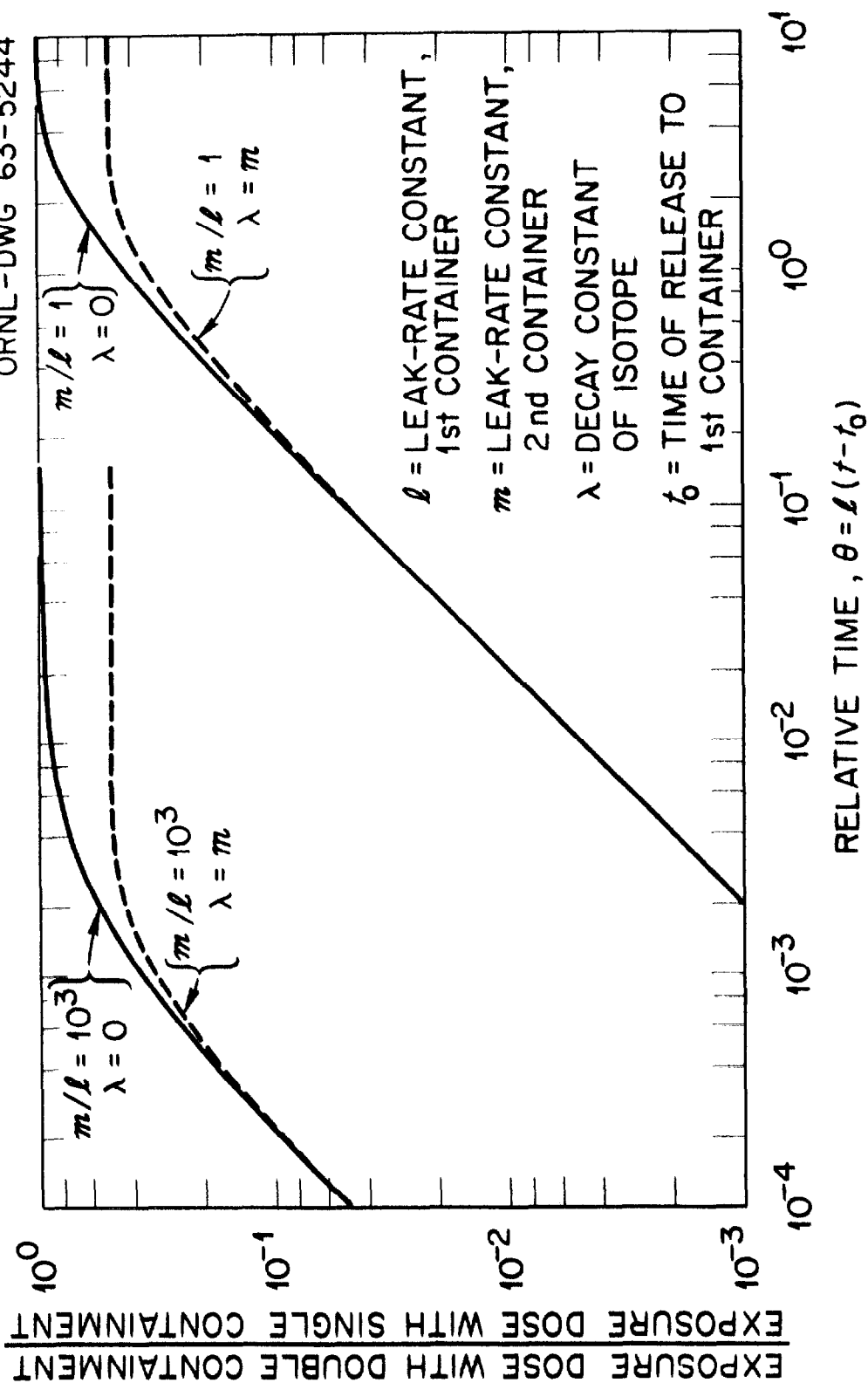


Fig. 5. Effect of Double Containment on Finite-Time Dose from Isotope Decaying Exponentially.

by radioactive decay of the isotope. Ultimately, however, the double-containment system gives no net reduction in dose if decay is negligible.

#### Application to the N.S. SAVANNAH

The N.S. SAVANNAH reactor system is enclosed in a conventional containment vessel that is, in turn, enclosed in the reactor compartment. The reactor compartment is maintained at slightly sub-atmospheric pressure by blowers that exhaust the compartment air through particulate filters and activated charcoal adsorbers.

The original purpose of the reactor compartment system was to reduce the release to the atmosphere of particulate fission products and radioiodine in the event of an accident. Further reductions in the release of all airborne material are provided by the holdup effect of the reactor compartment, although this effect has not been included in AEC safety justifications.

The noble-gas dose reduction resulting from compartment holdup was investigated using three different compartment exhaust rates, and the results are shown in Fig. 6. The data of Fig. 6 were calculated using Eq. (13). It was assumed that the gases would be released from the core instantaneously at the start of the accident ( $t_0 = 0$  in Eq. 13). The time-dependent amounts of the noble gases were calculated on the basis of reactor operation at full power for one year.<sup>2</sup> To account for possible non-uniform mixing in the reactor compartment, the compartment leak-rate constant, given by

$$m = \frac{\text{ventilation rate}}{\text{volume ventilated}},$$

was computed using approximately one-half the actual compartment volume. Figure 6 is valid for any primary containment leak-rate constant,  $l$ , in the range  $0 < l \leq 0.025 \text{ day}^{-1}$ .

During initial operation of the N.S. SAVANNAH the emergency ventilation system was operated at an exhaust rate of 200 ft<sup>3</sup>/min ( $m = 9.6 \text{ day}^{-1}$ ). At this exhaust rate the 2-hour dose from noble gases released to the atmosphere would be reduced 71% because of compartment holdup. The ultimate reduction in noble gas dose would be 20%. As can be seen from Fig. 6, a further reduction in noble gas dose could be effected by reducing the exhaust rate. At an exhaust rate of 40 ft<sup>3</sup>/min the 2-hour dose would be reduced 93%, and the ultimate dose would be reduced 50%. There is, of course, a lower limit on the exhaust rate because the reactor compartment must be maintained at sub-atmospheric pressure to assure effective operation of the filters and charcoal adsorbers. The results of this analysis show, however, that it would be desirable to reduce the compartment exhaust rate to the lowest possible value consistent with the negative pressure requirement.

#### References

1. Truman D. Anderson, "Time-Dependent Release of Gaseous Material from a Three-Barrier Reactor Containment System," USAEC Report ORNL-TM-397, Oak Ridge National Laboratory, November 16, 1962.

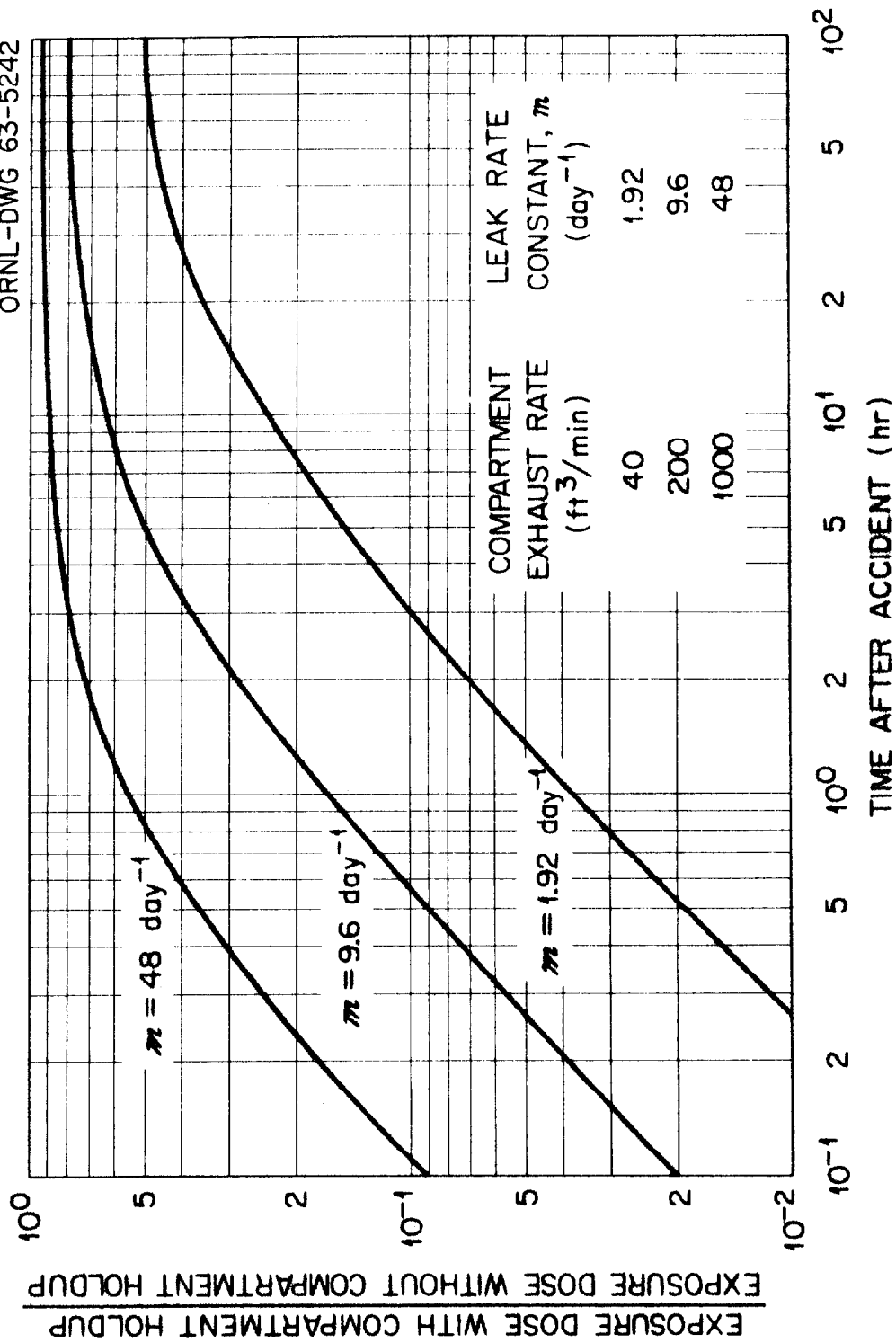


Fig. 6. Effect of Reactor Compartment Holdup on Dose from Noble Gases, N.S. SAVANNAH Reactor.

2. T. D. Anderson, J. R. Buchanan, W. B. Cottrell, M. H. Fontana, O. H. Klepper, and H. C. McCurdy, "Activity Release from the N.S. SAVANNAH in the Maximum Credible Accident," USAEC Report ORNL-3361, Oak Ridge National Laboratory, October 1, 1963.

#### DISCUSSION AND COMMENT

On the L and M values perfect diffusion is assumed within the containment vessel. In calculating, in particular, the M, one would do that by taking the flow rate out and dividing by the volume, and assume complete mixing of whatever it is you are interested in in the container. But one can adjust this M, get a hypothetical M, if experimental data were available, to show that there were not complete mixing.

Session Chairman: The Savannah River Plant will be represented by Mr. L. M. Arnett and Mr. B. C. Rusche with the presentation, "The Application of Iodine Absorber Units to the HWCTR Containment System." Mr. Arnett will deliver the paper.

THE APPLICATION OF IODINE ABSORBER  
UNITS TO THE HWCTR  
CONTAINMENT SYSTEM

by

L. M. Arnett and B. C. Rusche  
Savannah River Laboratory

This report is an analysis of the extent to which the installation of iodine absorber units in the containment building of the Heavy Water Components Test Reactor (HWCTR) can be expected to reduce radiation exposures of personnel in the vicinity of the facility during a release of fission products from the reactor core.

Because the radioiodines contribute substantially to the whole body dose and constitute almost entirely the thyroid dose, any mechanism that eliminates these isotopes from the materials escaping from the building reduces considerably the radiation hazard outside the building. The recent development of highly efficient iodine absorbers<sup>(1)</sup> made consideration of their installation in HWCTR a logical step.

#### SUMMARY

This evaluation of the maximum credible accident reveals that the operation of efficient iodine absorbers reduces the radiation hazards to commonly acceptable levels even though the leak rates from the containment building are relatively high. At leak rates in the vicinity of 2% of the building contents per day, the whole body dose and the thyroid dose received by personnel at the Plant boundary three miles from the HWCTR will be less than 25 r and 30 rem, respectively, in 30 days. The effect of the iodine absorbers is to reduce the expected whole body dose by a factor of two and the expected thyroid dose by a factor of one hundred.

#### DETAILS

##### 1. CONTAINMENT BUILDING

###### 1.1 Brief Description of Building

The containment building, which houses the reactor and principal auxiliary equipment, is designed to withstand an internal pressure of 24 psig. This is slightly greater than the equilibrium pressure that would result from the maximum credible accident, which is a rupture of the pressurized system full of D<sub>2</sub>O at an average temperature of 285°C. The maximum credible accident is defined in Section 2.1.

The building is 70 ft in diameter and 125 ft high; approximately half of the building is above grade. The enclosed volume is 420,000 ft<sup>3</sup> and the free volume is 320,000 ft<sup>3</sup>.

The above-grade portion of the building is carbon steel (ASME Spec SA-212, Grade B); the remainder is prestressed, reinforced concrete. The steel shell of the building is anchored to the concrete substructure by high strength bolts that are tensioned to compress the joint. The building was constructed in accordance with interpretations of the 1956 ASME Boiler and Pressure Vessel Code, Section VIII, "Unfired Pressure Vessels".

Major structural characteristics of the building are listed in Table I.

Table I  
STRENGTH OF THE CONTAINMENT BUILDING

	<u>Internal Pressure, psig</u>
Yield strength of steel shell	50
Yield strength of anchor bolts	63
Zero compression on main joint	50
Guaranteed <sup>(a)</sup> ultimate strength in girth cables used to prestress concrete	58
Minor cracks in concrete substructure	30

<sup>(a)</sup> Guaranteed by the manufacturer.

The building is designed to withstand an external pressure of 0.75 psig. A vacuum breaker prevents external pressure exceeding internal pressure by more than 0.25 psi. No relief is provided for internal pressure.

There are four floors in the building. The main floor at grade level is a single operating area. The minus 16-ft level is divided into two compartments, the minus 37-ft level into two compartments, and the minus 52-ft level into four compartments.

Building compartments are interconnected to equalize internal pressures. A 14-ft by 19-ft stairwell connects all levels of the building. Grating floors are used where structural and shielding requirements permit. The volume of the building is accessible to gases or vapors released from the reactor system.

Two air locks provide access to the building. One is reserved for emergency use. The normal access air lock, 10 ft in diameter and 16 ft long, contains two doors, both sealed by mechanically loaded gaskets. Interlocks prevent opening both doors at the same time. The emergency air lock is similar but is smaller.

Equipment is moved into and out of the building through a 7-ft wide and 7-ft high opening. During reactor operation this is sealed by a gasketed and bolted steel door.

Pressure-tight seals are provided where the building is penetrated by service piping and conduit. The largest penetrations are for the two air ducts of 24-inch diameter, one for fresh air, the other for exhaust air. These ducts are welded to the building shell. Each duct contains two air-operated butterfly valves that close in approximately 0.8 sec on signal from the control building. Service lines, through which any building pressure could be relieved, contain isolation valves that are operated from the control building. Electrical conduits enter the building through the substructure and terminate in junction boxes. A leakproof seal is installed in each conduit run.

### 1.2 Iodine Absorbers

Four iodine absorber units are provided within the building to remove radioactive iodine vapor released from ruptures of the process system. The units are installed in the following locations:

<u>No. of Units</u>	<u>Level</u>	<u>Location</u>
2	Zero	Platform at +22 ft
1	-16 ft	Right pump room
1	-37 ft	Left generator room

Each absorber unit, consisting of a moisture separator, activated charcoal filter, and fan, will filter 1000 cfm of building air. Provision is made on each unit for placing two inches of lead shielding across its top. The absorber fans will start automatically on a building "isolation" signal. Manual "Stop-Start" buttons and operating signal lights are provided in the control room. The fan motors are on emergency power and can start and operate under a pressure of 24 psig, a temperature of 108°C, and during operation of the deluge spray. Separate circuits in separate conduits run from the electrical substation to each fan motor.

The absorber units are located and oriented to secure maximum mixing of the building air. Six 1000-cfm portable fans, which operate continuously, assist in mixing the air. The motors of these fans are on emergency power and can operate at 24 psig, at 108°C, and during deluge spray.

The entrainment separators in the absorber units will remove 99% of the entrained liquid in steam-air mixtures drawn through the unit as determined by experiments reported in Reference 2.

The carbon filter cells in the absorber unit are activated coconut shell charcoal packed to a density of 35 lb/ft<sup>3</sup>. Each filter cell contains 56 lb of activated charcoal with an effective filter area of 14.6 ft<sup>2</sup>. Tests have shown that under the anticipated operating conditions of the absorber, iodine is removed with an efficiency of 99.99%.<sup>(1)</sup>

The absorbers are designed as integral units to facilitate removal and replacement, thus minimizing exposure to personnel. Construction features prevent recirculation of air within the unit and prevent bypassing the charcoal filter. Gasket and adhesives are designed to maintain the integrity of the unit for a minimum of eight hours at 110°C.

## 2. COURSE OF ACCIDENT

### 2.1 Definition of the Maximum Credible Accident

The maximum credible accident is defined as follows. A major break occurs in the high-pressure D<sub>2</sub>O system. The reactor core boils dry; the uranium metal in the test elements melts, breaks through the zirconium cladding, and ultimately reacts chemically with residual steam or air. The zirconium-uranium alloy in the driver tubes also reacts with steam and releases fission products. The hot D<sub>2</sub>O flashes through the break, and the temperature and pressure of the containment building increase. This is followed by a slower migration of fission product activity through the break into the containment building. Finally, the pressure and temperature inside the containment building are reduced by the action of the H<sub>2</sub>O spray system and by heat losses to the building structure and to the outside air.

### 2.2 Initial Conditions

The accident is assumed to occur after the reactor has operated for 365 days at a power of 70 MW. The average D<sub>2</sub>O temperature is 285°C. The accident starts when the system breaks and D<sub>2</sub>O flashes into the containment building at an assumed rate of 10,000 gpm. When 60 gal of D<sub>2</sub>O has been lost (less than 1 sec), a scram is initiated by an 8% decrease in the reactor pressure. The containment building is evacuated (as for all scrams). The safety rods will be completely inserted about 4 sec after the break. About 3 sec after the break the reactor pressure will be reduced enough for boiling to start in the hottest portion of the system. Boiling will begin in the reactor effluent lines and spread to the bulk moderator and coolant channels as D<sub>2</sub>O is lost. In one minute most of the D<sub>2</sub>O will be ejected.

### 2.3 Core Temperature Rise

Once the core has boiled dry, the temperature of the fuel will increase rapidly due to the fission product afterheat. Upper estimates of the temperatures that might be reached were made with the assumptions that the only mechanism for heat loss is radiation and that the powers in the driver and test fuel are 2.2 MW and 1.5 MW per assembly, respectively.

The calculations were carried out as follows. Heat from the fuel radiates to the housings. Two-thirds of the housing surface of the inner six test elements radiates heat to other test element housings, and one-third radiates to the driver ring. One-third of the surface of the housings for the outer six test elements radiates to other test elements, and two-thirds radiates to the driver ring. The outer half of the driver housings radiates to the thermal shield. The small amount of heat conducted around the periphery of the housing tubes was taken into account. It was assumed that the core boiled dry 10 sec after the break. The entire core would not be dry at this time, but some of the fuel channels probably would. The initial temperature of the driver fuel was taken as 400°C, that of the test fuel as 700°C, and that of all housing tubes as 300°C. Table II lists some of the parameters used in the calculation.

The maximum temperature in the driver fuel is reached in two minutes and is about 1750°C, or 100°C below the melting point of zirconium.

The temperature of the uranium metal test fuel, on the other hand, reaches the melting point of uranium (1160°C) in one minute. The molten uranium is expected to break through the zirconium cladding and flow to the bottom of the test assemblies.

Table II  
PARAMETERS IN CALCULATION OF CORE TEMPERATURES

<u>Element</u>	<u>Heat Capacity, MW sec/(°C)(ft)</u>	<u>Area x Emissivity, MW/(ft)(°K)<sup>4</sup></u>
Test fuel tube (thick-walled metal)	$1.0 \times 10^{-3}$	$1.0 \times 10^{-15}$
Test element housing	$0.15 \times 10^{-3}$	$1.4 \times 10^{-15}$
Driver fuel tube	$0.47 \times 10^{-3}$	$1.1 \times 10^{-15}$
Driver housing	$0.15 \times 10^{-3}$	$1.4 \times 10^{-15}$

At the elevated temperatures that will exist in the core, the zirconium and uranium will react chemically with residual steam. Tests on zirconium-steam reactions at temperatures up to 1750°C indicate that the reaction rate will be 1 to 5 mils/minute.<sup>(3)</sup> Thus, the driver tubes could be consumed in 15 minutes to one hour. The molten cores of the metallic uranium elements in the core will react more rapidly, but explosive reactions are not expected below 2200°C.<sup>(4)</sup> For the maximum credible accident, it is assumed that all of the uranium and zirconium in the fuel tubes react completely with water.

### 2.4 Heat Release

The heat released to the building consists primarily of the heat contained in 41,000 lb of liquid D<sub>2</sub>O at 285°C. This heat content is about  $7 \times 10^6$  pcu above that of water at 100°C, so this amount is available to boil the water into steam and heat the air in the building. The uranium-water reaction will add about  $0.6 \times 10^6$  pcu for 12 metallic test elements, and the zirconium-water reaction will add about  $1.5 \times 10^6$  pcu.

## 2.5 Pressure and Temperature Transients

The maximum values to which the pressure and temperature inside the containment building will rise were calculated from the equations that relate the heat available to the heat content of the components in the final state. In the calculations it was assumed that:

- the building is sealed completely as soon as the break occurs,
- all the air space in the building has the same temperature and steam content,
- heat transfer to the building and equipment is negligible, and
- the heat capacity of water vapor initially in the air and of helium from the reactor is negligible.

For the maximum credible accident, it is assumed that the chemical reaction is too slow to contribute sensibly to the maximum pressure in the building; thus  $Q$  is zero. The maximum pressure is 23 psig, and the maximum temperature is 108°C. The  $D_2O$  will flash out of the reactor in one or two minutes. The temperature and the partial pressure of steam will become fairly uniform throughout the containment building in a few minutes by spread of liquid  $D_2O$  and by convection currents set up in the air space. It is expected that the maximum values of the temperature and pressure will be reached in less than 10 minutes after the break occurs.

The maximum pressure is nearly equal to the 24-psig design pressure for the containment building, but is well below the 29 psig at which the building was test pneumatically. Even if the heat of reaction of the uranium and zirconium with water is immediately available ( $Q = 2.1 \times 10^6$  pcu), the maximum pressure will be 28 psi.

After the accident, the pressure and temperature inside the building are reduced by two main mechanisms: (1) the liquid  $D_2O$  transfers its heat to the building structure and equipment, and (2) the  $H_2O$  spray system is started by a high pressure signal to cool the air and condense the  $D_2O$  vapor.

Figure 1 shows the calculated pressure as a function of time after the spray is started. The spray flow is 1000 gpm, and it is assumed that the  $H_2O$ , the  $D_2O$  vapor, and the air are in equilibrium at all times. The spray alone will reduce the pressure from 23 to 10 psig and the temperature from 108° to 80°C in 15 minutes.

After the spray has stopped, the pressure and temperature will continue to drop slowly as heat is transferred to the building and equipment. It is estimated that 12 hours after the accident the pressure will be about 3 psig, and the temperature will be about 50°C. Additional  $H_2O$  can be supplied to the spray system from the wells at 120 gpm.

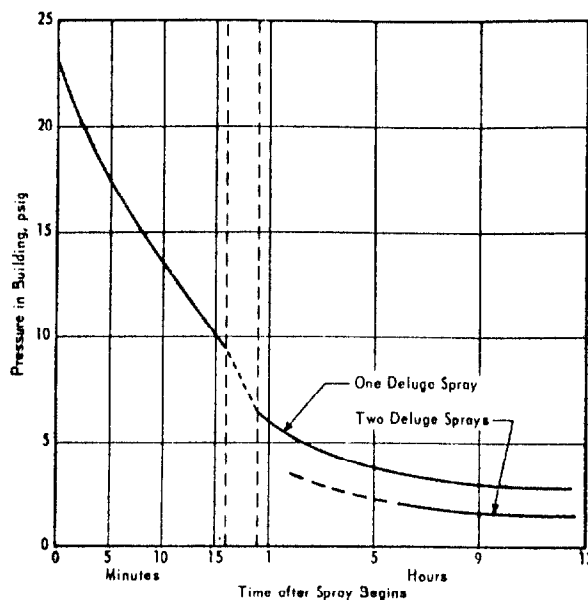


FIG. 1 - PRESSURE REDUCTION BY SPRAY SYSTEM

### 3. RADIATION HAZARD

#### 3.1 Basis of Hazard Evaluation

The calculated dose for all radiation hazards is based on a fission product inventory equivalent to operation for 365 days at 70 MW. The accumulated exposure is about 25,000 MWD. Reactivity will limit exposure for the first few charges to about 15,000 MWD, but subsequent charges may operate up to 25,000 MWD exposures.

The commonly accepted assumptions and techniques<sup>(5)</sup> were used to calculate the doses for ingested radiation hazards. Similar assumptions, but slightly different techniques, were used to calculate the whole body dose from the cloud.<sup>(6)</sup> In both cases the dispersion of radioactivity in the atmosphere was estimated from the Sutton equation for a continuous point source at ground level. Therefore, many of the assumptions and parameters are common to both calculations.

The common form of the Sutton equation is:

$$C' = \frac{2Q}{\pi C_y C_z V D^{2-n} T} \exp \left[ -\frac{Y^2}{C_y^2} - \frac{Z^2}{C_z^2} \right] \left[ \frac{1}{D^{2-n}} \right]$$

where

- $C'$  = concentration of radioactive material in the air, curies/meter<sup>3</sup>
- $Q/T$  = release rate of radioactive material, curies/hour
- $n$  = turbulence index
- $C_y$  = dispersion coefficient in the horizontal plane, meters <sup>$n/2$</sup>
- $C_z$  = dispersion coefficient in vertical plane, meters <sup>$n/2$</sup>
- $V$  = wind velocity, meters/hour

D = downwind distance from source, meters  
 Y = crosswind distance from cloud axis, meters  
 Z = vertical distance from cloud axis, meters

The dose from a ground level cloud can be obtained with appropriate forms of the equation. The parameters used in these calculations for inversion conditions are given in Table III and are typical of those used in Reference 5.

Table III

ATMOSPHERIC DISPERSION PARAMETERS

<u>Parameters</u>	<u>Value</u>
n	0.5
C <sub>y</sub>	0.4 meters <sup><math>\frac{1}{4}</math></sup>
C <sub>z</sub>	0.07 meters <sup><math>\frac{1}{4}</math></sup>
V	3200 meters/hour

The following assumptions were made concerning the release, leakage, and dispersion of fission products:

- a. The maximum credible accident releases to the building the following percentages of the fission product inventory:
 

noble gases	100% ( $1.25 \times 10^7$ curies)
halogens	50% ( $6 \times 10^6$ curies)
solid products	1% ( $3 \times 10^4$ curies Sr)
equivalent gross fission product release	15%
- b. Fifty percent of the iodines in the containment vessel remain available for release to the atmosphere. The other 50% is removed by absorption, settling, or adherence to internal surfaces. As the building atmosphere is quite humid after operation of the deluge system and release of steam, the moisture serves as a trap for the iodine. This assumption is conservative.
- c. The internal iodine absorbers described in Section 1.2 remove the iodine with an efficiency greater than 99%,<sup>(1)</sup> at an effective removal rate of  $5 \times 10^{-5}$  sec<sup>-1</sup>. This assumption is based on 1000 cfm circulation.
- d. The release rate from the containment building is constant for the duration of exposure. The deluge system will reduce the building pressure to 10 psig before the fuel melts 15 minutes after the break. In 12 hours the building pressure is calculated to be less than 3 psig.
- e. The wind direction does not shift for the duration of exposure. Since the wind direction at Augusta, Ga., in 1957 was never constant for as long as 24 hours, this assumption is conservative.
- f. Atmospheric dispersion occurs under severe inversion conditions which persist for the course of the accident. Since measurements at Charleston, S. C., and Atlanta, Ga., indicate that the maximum frequency of day-long inversions is about 60% in June and 10% in January, this assumption is conservative.
- g. Cloud depletion or fallout does not occur during cloud travel. This assumption is conservative.
- h. Fission products do not decay in transit. Decay and removal by filtration is assumed while the fission product mixture is in the building.

1. Only the gamma activity contributes to the whole body dose from the cloud. The cloud is assumed to be large with a uniform source distribution. The absorption coefficient ( $\mu$ ) for the air is  $0.00003 \text{ cm}^{-1}$ .

### 3.2 Direct Radiation from Building

If all the radioactivity listed in Section 3.1 is released in fifteen minutes, the radiation dose rate in the vicinity of the containment building will be as shown in Table IV.

Table IV

Distance from Bldg, ft	RADIATION FOLLOWING MAXIMUM CREDIBLE ACCIDENT							
	Dose Rate, r/hr, at the Indicated Time after the Accident							
	15 min	1 hr	3 hr	10 hr	1 day	3 days	10 days	30 days
330	620	360	120	48	32	12	7.2	0.80
660	160	84	24	8.0	5.0	2.0	1.2	0.08
1300	13	7.2	2.0	0.80	0.40	0.16	0.08	0.01
2000	1.8	1.2	0.32	0.08	0.04	0.02	0.01	--
3300	0.07	0.04	0.01	0.01	--	--	--	--

### 3.3 Thyroid Dose from Cloud

The release rate of radioactive isotopes to the atmosphere is calculated from the equation

$$Q(t) = \frac{1}{P} \left[ \frac{\lambda_1 F q}{\lambda_1 + \lambda_R + \lambda_f} \right] \left[ 1 - e^{-(\lambda_1 + \lambda_R + \lambda_f) t} \right]$$

where

- $Q(t)$  = the activity from a specific isotope released from the building to the atmosphere during the interval  $t$ , curies/MW
- $q$  = the reactor inventory of a specific isotope, curies
- $P$  = reactor power, MW
- $F$  = fraction of a specific isotope available for release from the building
- $\lambda_1$  = release rate from building,  $\text{sec}^{-1}$
- $\lambda_R$  = isotope decay constant,  $\text{sec}^{-1}$
- $\lambda_f$  = removal rate by the iodine absorbers,  $\text{sec}^{-1}$
- $t$  = time, sec

The  $Q(t)$  is the source term for the atmospheric dispersion equation that is derived from the Sutton equation

$$A(t) = \frac{2R Q(t) P}{A C_y C_z D^{1.5}}$$

where

- $A(t)$  = amount of radioactivity inhaled from the cloud during an exposure of  $t$  seconds, curies
- $R$  = breathing rate,  $\text{meters}^3/\text{sec}$

Other terms are the same as defined previously.

The thyroid dose was calculated as the sum of the doses from the isotopes I-131, I-132, I-133, I-134, and I-135. The values of the parameters used were the same as for the whole body doses except as listed below.

Parameter	Value
F	0.25
q	Saturated activities at reactor power = 70 MW
$\lambda_f$	$5 \times 10^{-5} \text{ sec}^{-1}$ (this corresponds to a circulation rate of 1000 cfm in a building volume of 320,000 ft <sup>3</sup> )
t	Infinity (i.e., until all iodine isotopes have decayed)
R	20 meters <sup>3</sup> /day = $2.3 \times 10^{-4} \text{ meters}^3/\text{sec}$

Calculations, on the above basis, show that the thyroid dose will be less than 30 rem at the Plant boundary, which is three miles from the reactor, for leaks of less than 3% of the building contents per day. This dose is less by a factor of 10 than the permissible dose of 300 rem and is less by a factor of 100 than the dose that would be experienced if the iodine absorbers were not installed. When maintenance is adequate to ensure operability of the iodine absorbers and to minimize the leakage from the building, the hazard from the thyroid dose is largely eliminated.

### 3.4 Whole Body Dose from Cloud

The whole body dose received by a person who remains at the centerline of the cloud for 30 days was estimated from the following equation, which was derived from the Sutton equation for the atmospheric condition described in Section 3.1.

$$r = \frac{1.84 \times 10^5 p f a t}{v D^{1.5}}$$

where

- r = dose, roentgen
- p = reactor power, MW
- f = release rate, fraction/hour
- a = fraction of gamma power from noniodine fission products averaged over the exposure time
- t = exposure time, hours
- v = wind velocity, miles/hour
- D = downwind distance, miles

The values of the parameters used in the calculations were

- t = 720 hours (30 days)
- a =  $5.1 \times 10^{-4}$
- p = 70 MW
- v = 2 miles/hour

Calculations on the above basis show that the dose will be less than 25 r in 30 days at the Plant boundary for average leakage rates of less than 1.6% of the building contents per day.

The evaluation of the hazards from both the whole body dose and the thyroid dose shows that relatively large leakage rates are permissible. In fact, leakage rates as high as these persisting for 30 days are not physically attainable when the leakage of significant amounts of fission products begins at an initial pressure of 5 psig. After a leakage of 25% of the building contents, the pressure in the building is zero psig, and this leakage corresponds to a leakage rate of 0.83%/day for a total of 30 days. Therefore, the calculation of a permissible leakage rate greater than 0.83%/day for a period of 30 days implies that there is no limit on the leakage rate. Obviously, this implication is not correct since at high leakage rates the assumptions in the above calculations are not valid. If the bulk of the fission products were to escape in a short period after the maximum credible accident, the average activity of the fission

products would be greater than the activity averaged over 30 days, and in an extreme case the iodine absorbers would not have time enough to remove an appreciable fraction of the iodine.

The assumptions used in the calculations of the whole body dose were modified to permit the calculation of a physically more meaningful limit on the leakage rate. Assumption d in Section 3.1 was modified to: the release rate from the containment building is constant for the duration of the exposure, and the integrated leakage is 25% of the building contents. Calculations with this modified assumption show that the dose at the Plant boundary would be 25 r for a leakage rate of 4.5%/day lasting for a period of 5.6 days. The use of the modified assumptions has only a negligible effect on the thyroid dose, since the iodine absorbers remove iodine in a very short time compared to five days. This evaluation, therefore, shows that, even with very conservative assumptions, no exposures greater than 25-r whole body dose and 300-rem thyroid dose will be expected beyond the Plant boundary if the leakage rate from the building is less than about 2% of the building contents per day when the initial pressure is 5 psig.

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## DISCUSSION AND COMMENT

While the material is still in the reactor, we assume for each isotope, 132, 133, 134 and 135, that while it is in the building, it decays by natural radioactive decay, and we also assume that the iodine absorbers were removing the iodine from the atmosphere at the circulation rate of 1,000 cfm through the filter, and we assume that that was 99% effective.

Although we do have four filters in the building, we assume that actually only one was operative. Of the 320,000 cubic feet of air in the building, 1,000 cfm, was being circulated through this absorber unit. So there are two processes for elimination of the iodine; radioactive decay and absorption in the iodine units.

We test halogen removal capabilities by a freon test. We circulate air to these units, and air entering the unit contains freon, and we then measure the freon that is present in the air that passes through the filter, and from this we get the efficiency.

The experimental results of atmospheric values we did use, we took directly from this reference that I cited, which is 14844, a document put out by the AEC Division of Licensing and Regulations in Washington, and it is entitled, "Calculation of Distance Factors for Power and Test Reactor Sites."

We are concerned about how well these charcoal filters will maintain their efficiency over a two or three year span, and we do have testing programs under way, and Mr. List indicates we will take out individual units and test them from time to time, so in this way we will accumulate data. We also do have some work in progress in which we would actually test the efficiency of these filtration absorption systems actually in place. The best we can say is that we are still working on the problem, and hope to accumulate data which will serve as the basis for giving a better estimate of what the lifetimes of these units are.

Session Chairman: Hanford will be represented by the contribution, "Hanford Experience with Reactor Confinement," a paper by J. W. Green, W. V. Thompson and H. Copeland. Mr. Green will present the contribution.

# HANFORD EXPERIENCE WITH REACTOR CONFINEMENT

by

J. W. Green, W. V. Thompson, and H. Copeland  
Hanford Atomic Products Operation

## INTRODUCTION

Hanford's experience with the reactor confinement system, in regards to its ability to perform the intended function, has been good. Although there have been no incidents since the installation of the confinement system wherein fission products would have been released to the atmosphere without a confinement system, extensive tests are conducted routinely to assure that the system will contain such fission products if called upon to do so. Before going into the details of these tests and the operational history of the various segments of the confinement systems, I would like to describe briefly what comprises the confinement system on a Hanford reactor.

The typical layout, shown in Figure 1, graphically illustrates the basic features of the confinement systems. Each system is composed of:

1. A ventilation system to force the air from the various zones surrounding the reactor block, through the filters before exhausting it through the stack to the atmosphere.
2. A filter facility that contains the particulate and charcoal filters.
3. A sample building, constructed above the ventilation ducts leading to and from the filter building, containing the instrumentation for monitoring for excessive radioactivity.
4. A dense, finely atomized, spray system within the rear face enclosure.
5. A stack which takes the air stream high above ground before exhausting it to the atmosphere.

### I. VENTILATION SYSTEM

During normal operation, ventilation air is supplied to the building by two supply fans. A portion of this air is diverted to the confinement zones, while the rest of it is used in offices, shops, and storage areas within the reactor building. The air is then drawn into a common plenum which is connected to the suction side of the exhaust fans. Prior to installation of the confinement system, the fans discharged air directly to the stack without filtering. The plenum at the stack has been breeched, diverting the air flow through the newly constructed filter building before its release to the surrounding environment.

The confinement program improved the reliability of the existing ventilation system by modifying the existing exhaust fans and adding equipment to permit continuous flow of exhaust air from the confinement zone surrounding the reactor proper. A typical ventilation flow diagram

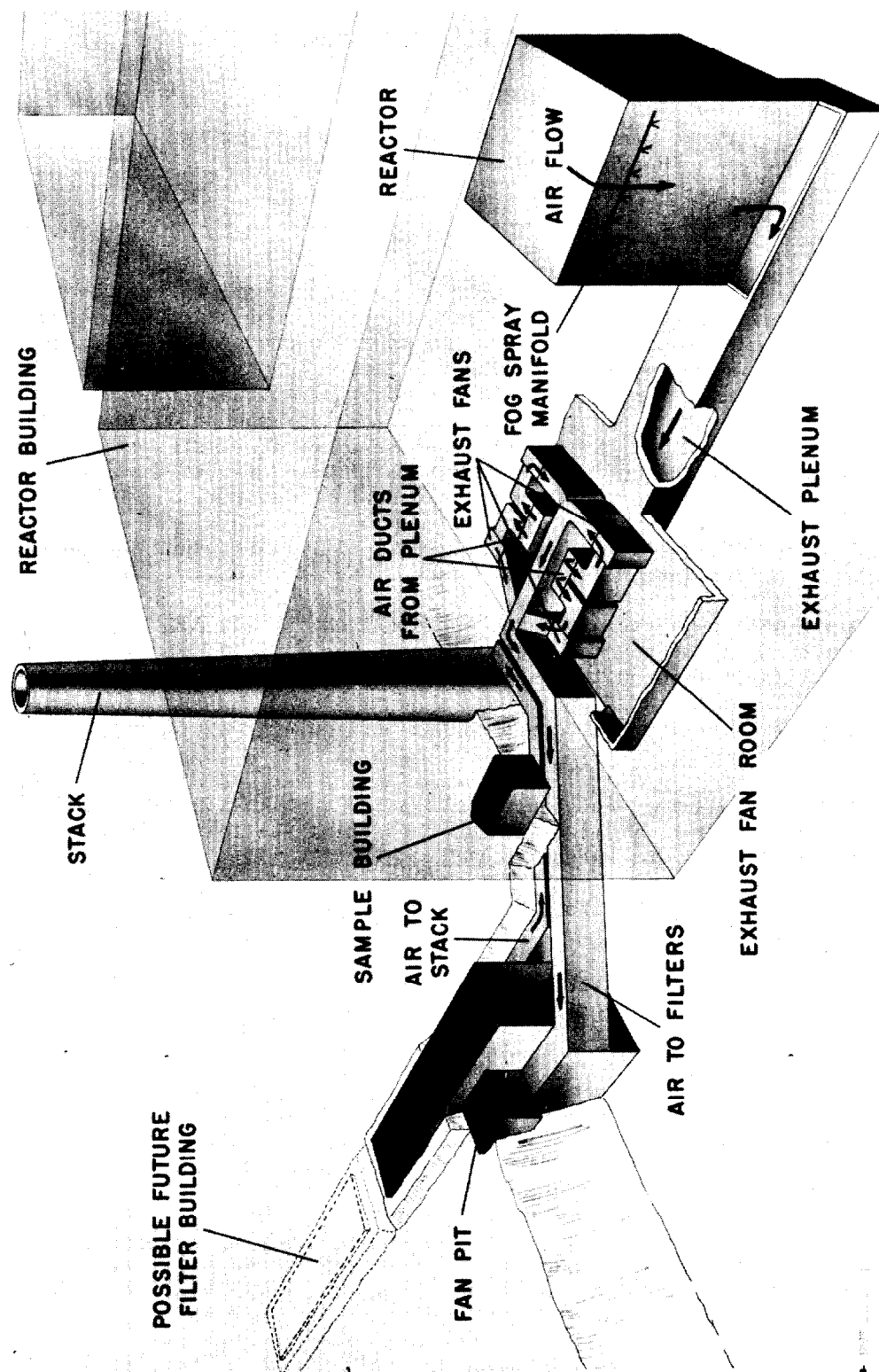


Figure 1 - FACILITY LAYOUT

is shown in the next slide (figure 2). The modifications to the system were as follows:

1. The electrically driven exhaust fans were upgraded to provide for the increased pressure drop of the new filter facility.
2. Emergency exhaust fans were provided with dependable steam or diesel engines in the event normal electrical power was lost.
3. Instrumentation was provided to automatically shut down the supply fans in the event of a reactor incident to insure that zone pressures are maintained negative with respect to atmospheric pressure.
4. Ventilation barriers were installed to permit the air balance crews to successfully balance the confinement zone pressures to slightly less than atmospheric pressure.
5. Alarms were provided to warn operating personnel of any significant change in ventilation balance.

Although the system functions well when responding to a simulated incident, certain problems have been experienced since the system has been in service.

#### Operating Experience

Pressures within the range of 0.02 to 0.06 inch water gage with respect to atmospheric reference within the confinement zone boundary have been impossible to maintain in a practical sense. Spurious trips of the confinement zone pressure alarms plague reactor daily operations. As yet, all causes of these spurious trips have not been fully resolved; however, programs are now underway to determine them. The reactor buildings are large and leakage rates high, which results in fluctuation of building pressures caused by varying wind velocities.

The automatic, spring loaded, valve installed on the emergency exhaust fan steam turbine drives failed to open in most instances after being turned over for operational use. The problems with this valve were solved by proper adjustment of the valve linkages, lubrication of the valve moving parts on a mandatory schedule and the installation of a strainer upstream of the valve. The operational experience gained by this type of valve in service indicates satisfactory performance is only obtained through special care and maintenance.

To ensure proper operation and performance of the equipment to maintain continuity of exhaust air from the reactor to the filters certain functional checks are mandatory.

#### System Testing

In order that the system will operate satisfactorily when required, these functional checks are made at intervals ranging from weekly to quarterly. Each week the ability of the emergency exhaust fans to assume the ventilation air load is checked, and the fans are run until bearing temperatures reach equilibrium. On a three month interval the overspeed trip mechanism on the emergency exhaust fans are checked for operability. The gravity vacuum relief damper which is installed in

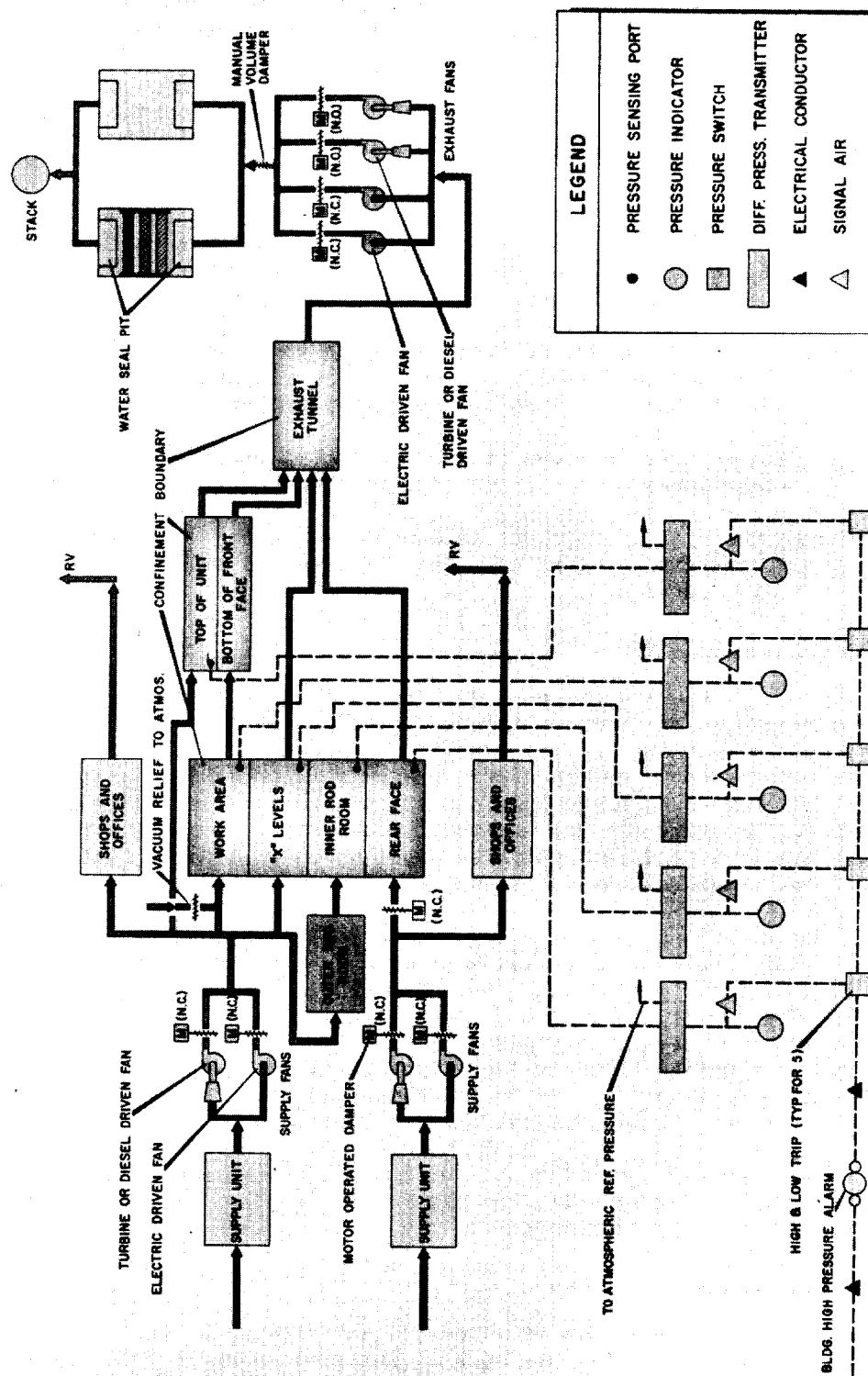


Figure 2 - TYPICAL FLOW DIAGRAM

the supply duct is manually checked for operability. This was installed to prevent collapse of the supply duct upon loss of supply air during an incident.

## II. FILTER FACILITY

The confinement program at Hanford constructed a filter building approximately 59 feet long, 39 feet wide and 35 feet high, shown on the next slide (figure 3). The building is of reinforced concrete construction and is almost entirely below ground. It is connected to the existing stack breeching by two underground concrete ducts and metal breeching transition ducts above grade. The concrete intake and exhaust plenums in this building are provided with turning vanes to help streamline the flow of air to the filter cell. This lower section of the duct also doubles as a water seal pit when the filter cell is isolated for maintenance. All of the interior concrete surfaces located below ground level have been coated with a polyvinyl coating to seal any cracks in the concrete that could possibly be a leakage point in the event these surfaces be subjected to contaminated air.

Although normal radiation activity on the filters is quite low, the underground installation was used since (1) earth shielding is inexpensive, (2) abandonment in place would be simpler should this become necessary and, (3) the building and associated duct work causes less hinderance to moving vehicles and personnel.

There are two cells in this filter building that are separated by an operating gallery. Each cell contains provisions for three banks of filters in series and each bank provides for two aluminum filter frames which hold from 24 to 40 individual hi-efficiency or activated charcoal filters rated at 1000 cfm each. The volume of the reactor exhaust air of the plants varies from 85,000 to 150,000 cfm. The buildings were equipped with 96 filters for the lesser air flow and 160 filters for the greater air flow.

As a result of filter life tests, it was found that roughing filters added little to the effectiveness of the system and were nearly as expensive as the absolute filters. Upon completion of the filter building, the absolute filters were placed in the first bank and the activated charcoal filters in the third bank. This left the second bank unused.

### Operational Experience

Installation of the absolute filters in the building started in early December, 1960 and was completed by late January, 1961. At this time the ventilation system was balanced; however, the final balance to confinement system requirements could not be completed until the activated charcoal filters were installed. The last activated charcoal filters were placed in service in October of 1961. These balances have proven to be tedious to make due to the magnitude of the system, the low pressure differential requirements within the zones and the fluctuations of building pressures caused by varying wind velocities. In April of 1962 the balancing of the system at each reactor plant was completed. The build-up on the filters since that time is shown in the next slide (figure 4). The two plants shown have 112 and 160 filters with average flow rates at 900 and 800 cfm and have a corresponding  $\Delta P$  build-up at 2.4 and 1.5 inches of water. Other Hanford plants are located between these two curves. The wide spread of the curves is a

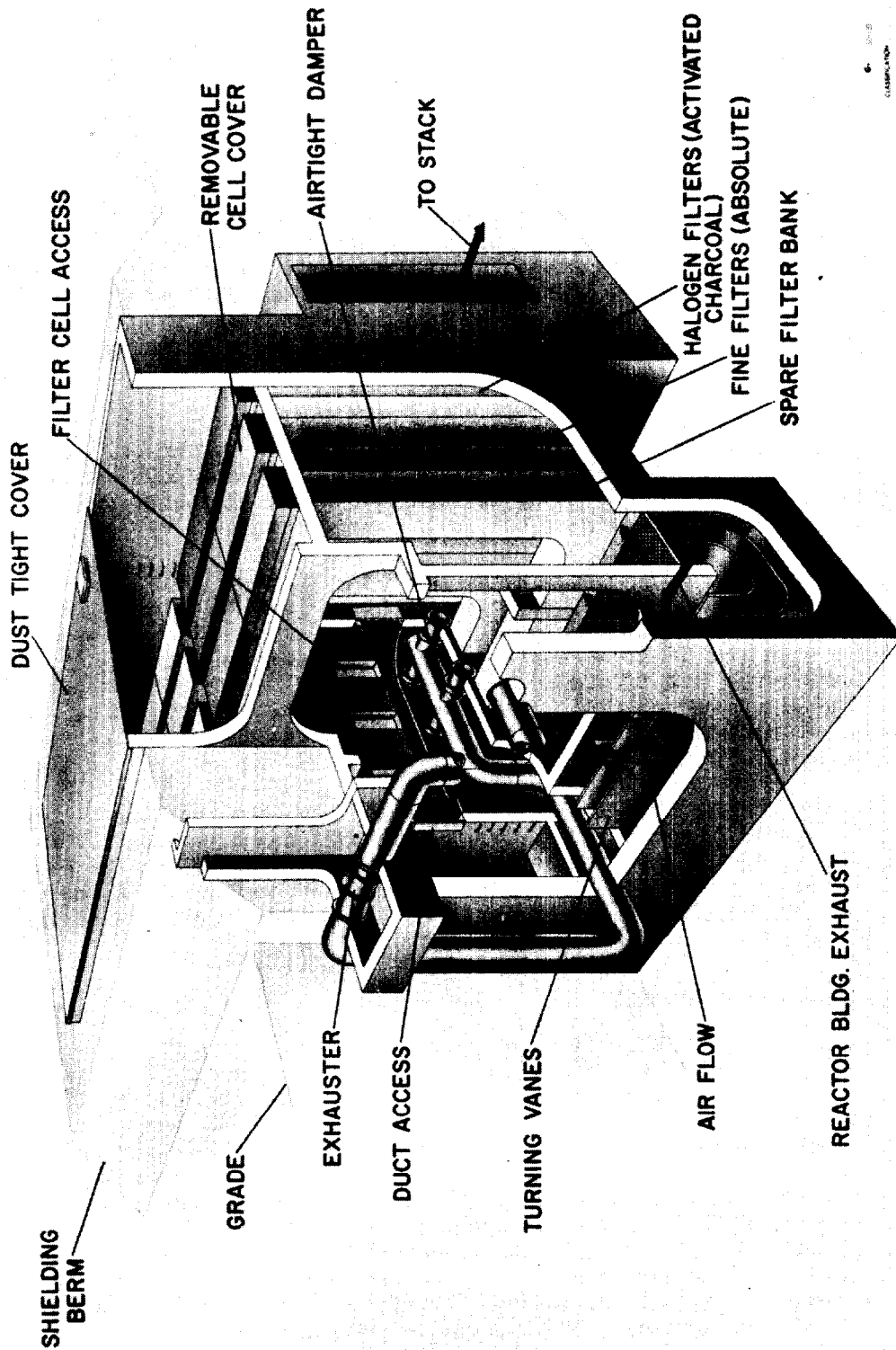


Figure 3 - FILTER BUILDING

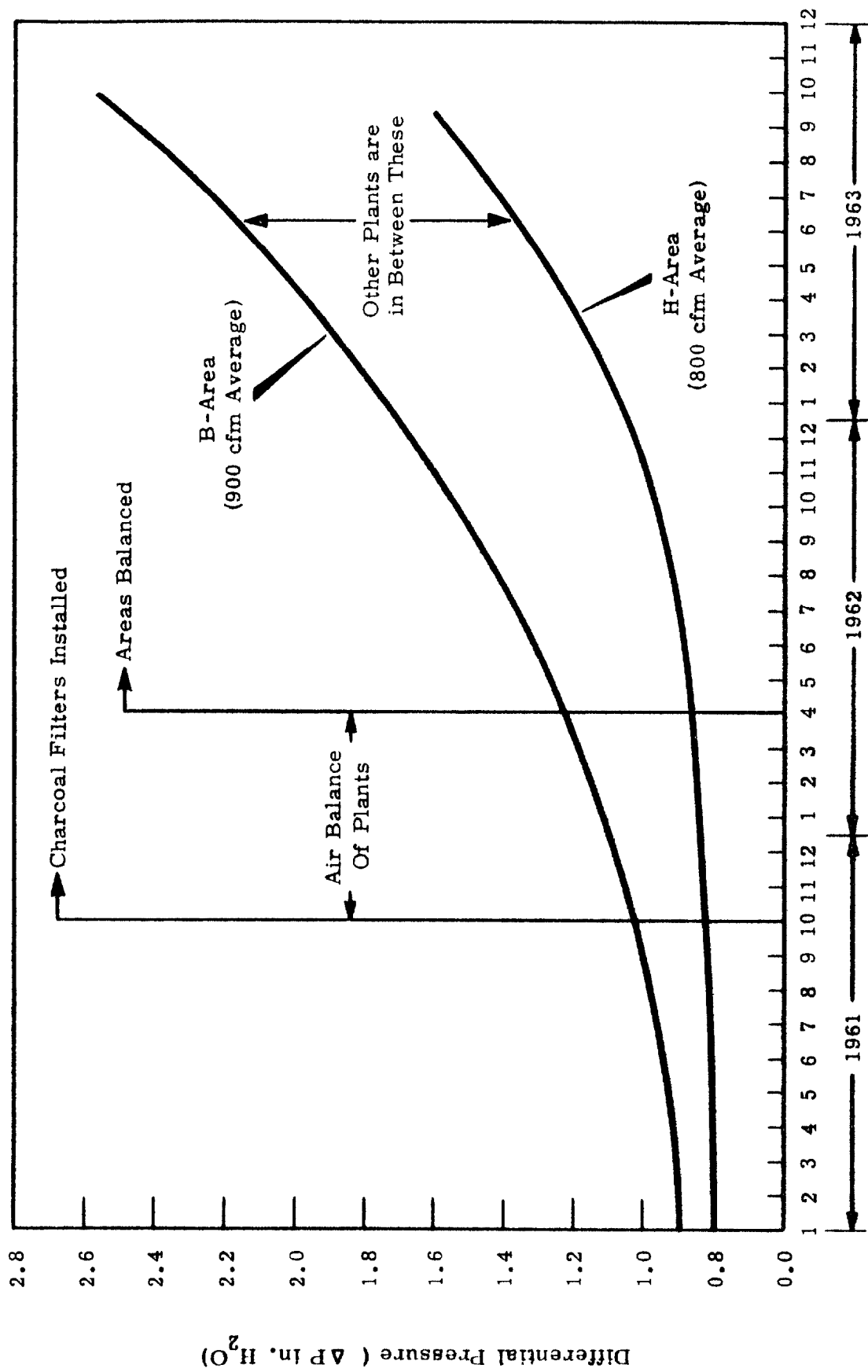


Figure 4 - LOADING HISTORY OF HIGH EFFICIENCY FILTERS (1961 - 63)

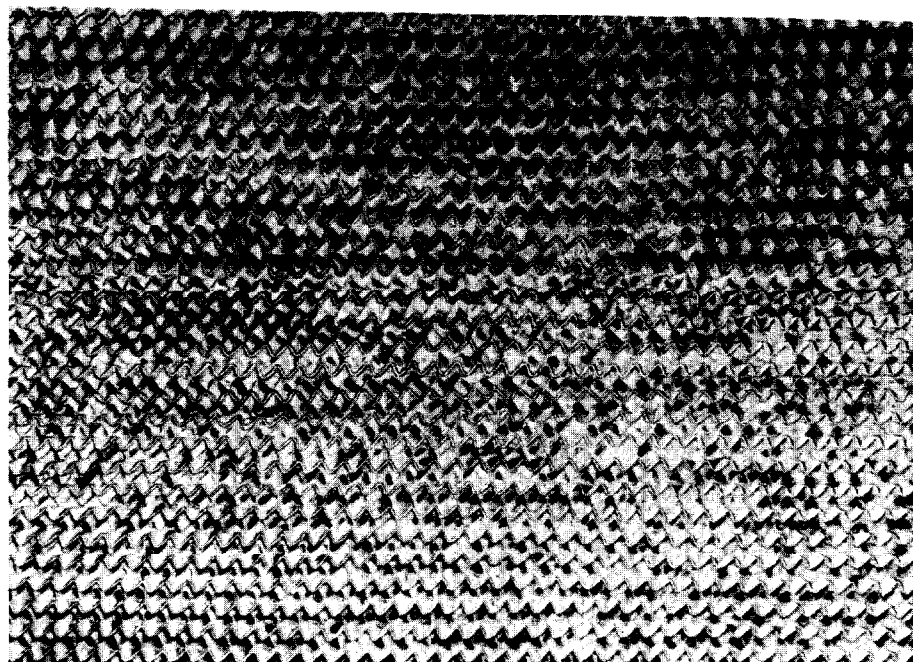
function of many variables which include heights of supply air inlet plenums from the ground, location from prevailing wind directions, particulate producing activity within the reactor building confinement boundary, control of closed doors, length and areas of exhaust air plenums, continuity of washing the supply air to the confinement zone, and preventive maintenance on the supply air filters. As mentioned before, the hi-efficiency filters have been in service prior to this time. These filters will experience three years, in January, 1964, in-line filtration. It is estimated that filter replacement will be required when the pressure drop across the filters reaches 2 to 4 inches water gage and appears that a likely replacement of some hi-efficiency filter banks will start somewhere between mid 1964 or early 1965. This indicates that the filters will exceed the anticipated three year life except for those filters that were replaced recently in the bottom rows in 11 out of a total 16 cells in the eight reactor plants.

During one of the plants routine removal of a charcoal filter test section, the hi-efficiency filter bank located upstream of the charcoal filter bank was visually examined for deterioration and damage. The examination disclosed that the bottom two rows of the absolute filter bank in this cell needed to be replaced. The next slide (figure 5) illustrates the damages to the first and second rows. As can be seen, the folding edge of the media has completely collapsed onto the inner separator and spread out between the outer separators. A closer look shows the inner separator to be completely exposed in many places on the bottom row. As a result of this examination the opposite filter cell in this plant and both filter cells in the other plants were inspected. As mentioned before, 11 out of 16 cells visually showed similar damage to either the bottom row or the bottom two rows. The actual phenomena that takes place is unknown but is believed to be the effect of water conveyed onto the filters. Referring back to the slide (figure 4) of the filter building, moisture in terms of humidity can get to the filters via the exhaust air stream, and the other possibility for water being carried to the filters exists when isolating a cell for entry. Upon entering the plants filter cells for visual inspection, it was discovered that in some plants the bottom two rows of filters were wet. Investigation, in each case, showed air flowing through the cell when the inlet seal pit was being filled with water. The result is water is blown onto the filters due to the greater air pressure on one side of the seal wall as compared to the water sealing heights on the other side. As a result of the investigation, the standard operating procedures for isolating cells have been revised to eliminate all possibilities of exhaust air flow through the cell while filling the inlet seal pit. Also, for further precautions the filters are being moved from the first stage to the second stage; thereby, giving additional distance between the filters and the seal pits.

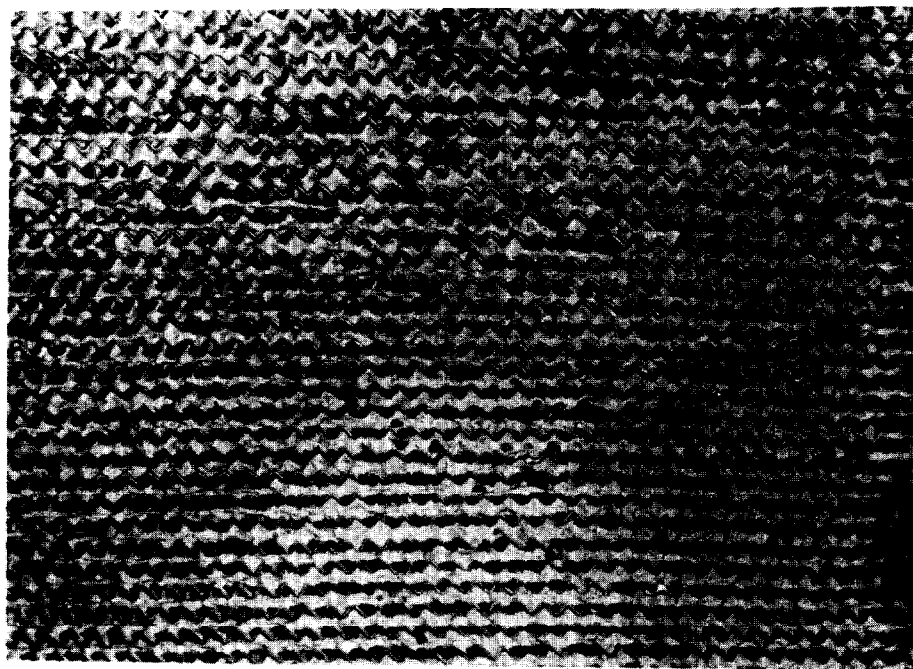
Until the bottom rows of the filters became damaged, engineering effort was directed towards providing a more reliable inflated neoprene filter frame seal on the halogen filter banks. Approximately half of the 56 seals in service have experienced leaks. In all instances the leak was located at the vulcanized joints. Recently, new seals have been purchased under more rigid specifications, and it is hoped that these seals will provide the expected three year life.

#### System Testing

To insure the high degree of filtration efficiency required of the confinement facilities the hi-efficiency filters are tested annually for leakage by in-place DOP testing and visually inspected quarterly for loading and damage. As mentioned before, each cell is supplied with



**BOTTOM ROW**



**SECOND ROW**

Figure 5 - DAMAGE ON LOWER ROWS - HIGH EFFICIENCY FILTERS

test sections (samples) which are representative of the in-service halogen charcoal filters. On an annual basis a sample is removed and tested to determine the absorptivity of the charcoal filters. These laboratory tests indicate the useful life of the filters. Further information on these charcoal filters is presented in Jerry McCormack's talk.

### III. INSTRUMENTATION

#### A. Reactor Zone Static Pressure System

1. Description (Slide No. 6) The static pressure in each of five ventilation zones in the reactor building is monitored with sensitive differential pressure instruments. Each zone is connected to the measuring tap of a low range pneumatic differential pressure transmitter. A common atmospheric reference line is connected to the reference tap of each transmitter. The pneumatic output signal from each transmitter goes to a static pressure indicating gage, and to a pressure switch. When the static pressure in any of the zones reaches an unsafe level the pressure switch contacts actuate a warning alarm. The zone static pressures, with respect to atmosphere, established by balancing the ventilation system are:

Inner Rod Room  $-.04$  In.  $H_2O$   
Work Area  $-.02$  In.  $H_2O$   
Experimental Level  $-.02$  In.  $H_2O$   
Top of Unit  $-.025$  In.  $H_2O$   
Discharge Area  $-.08$  In.  $H_2O$

2. Operating Experience During windy conditions the zone pressures fluctuate so as to actuate the warning alarm. Difficult has been experienced in locating the atmospheric reference to minimize the effects of wind direction changes.
3. System Tests The static pressure transmitters, indicating gages, and pressure switches are routinely calibrated. The zone static pressures are routinely checked and ventilation air balance adjustments made as necessary.

#### B. Gas Sampling and Radiation Monitoring System

1. Description The reactor building exhaust air is continuously monitored both upstream and downstream of the confinement filters for radioactive particulate matter, and for radioactive Iodine ( $I^{131}$ ). Each filter-cell also has a gamma monitor for measurement of accumulated radioactive build-up on the filters.

The particulate monitors draw a continuous air sample from the ventilation exhaust duct. The sample is passed through a continuously moving filter paper upon which any particulates are deposited. The filter paper is monitored with a beta sensitive radiation detector. The radiation pulses are counted with a log count rate meter and recorded on a six decade strip chart recorder which actuates an alarm upon high count rate.

The upstream and downstream radioactive Iodine ( $I^{131}$ ) monitors continuously monitor an air sample from the ventilation exhaust duct. The upstream sample is passed through a gas pig where a

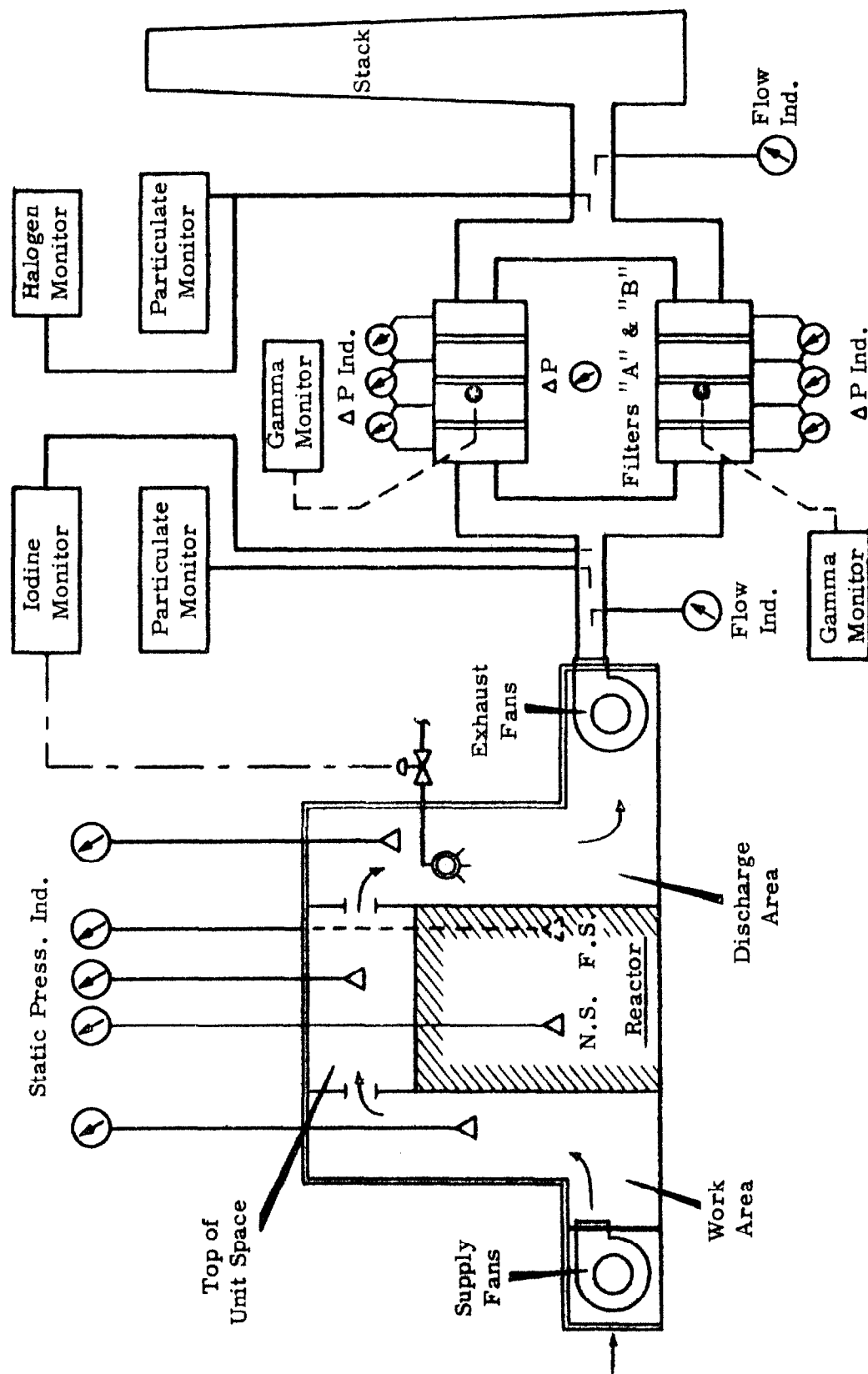


Figure 6 - CONFINEMENT SYSTEM

fixed volume is monitored for  $I^{131}$  with a scintillation type gamma detector. The downstream sample is passed through a charcoal filter within a pig where the accumulated  $I^{131}$  is monitored with another scintillation type gamma detector. The upstream pulses are amplified, discriminated for  $I^{131}$  energy level fed to a differential count rate meter and recorded on a strip chart recorder. The fog spray actuation and shut down of supply fans is initiated by a contact meter controller connected in parallel with the recorder. The downstream pulses are amplified, discriminated for  $I^{131}$  energy level, fed to a log count rate meter, and recorded on a strip chart recorder. Contacts in the recorder actuate a high activity alarm.

2. Operational Experience Considerable attention was required in setting up the instruments and performing the initial calibration. Filter paper drive failures have been experienced on the particulate monitors. Some overheating of electronic instruments required additional cooling. Day-to-day activity trends are noted on the recorders and changes correlated with processing activities in the reactor. No activity level has been high enough to actuate the fog spray system.
3. System Tests The response of these systems is routinely tested with radioactive sources. Initial response and calibration proof tests were run on the Iodine monitors using an Iodine generator. Routine test and calibration of the electronic instrumentation is also performed.

#### C. Ventilation Flow and Filter Differential Pressure

1. Description A pitot tube is located in both the inlet and outlet ventilation ducts of the filter building. The pitot tubes are connected to flow indicating gages.

The filter banks in each filter cell have upstream and downstream pressure taps connected to gages that indicate the pressure drop across each filter. Also, the total pressure drop across each filter cell is indicated on differential pressure gages.

No abnormal performance of pitot tubes or gages has been experienced. The gages are routinely calibrated for accuracy.

#### IV. FOG SPRAY

The fog spray is a finely divided water spray system located within the rear face (fuel discharge) enclosure and is designed for automatic or manual operation. This is shown in more detail on the next slide (figure 7). Basically the system is a four inch stainless steel header mounted in the center of the ceiling in the rear face enclosure with spray nozzles attached at intervals that provide for optimum coverage by the water spray. When the system is in the automatic mode of operation, the valving to the fog sprays is activated by the scintillation detector system which continuously monitors the reactor building exhaust air. The manual operation of the fog spray is accomplished by either manual operation of valves at the valving station or by turning the fog spray switch located in the control room to the "emergency on" position which in turn operates the valves supplying water to the fog spray system.

Specifically the system is designed for the multiple purpose of:

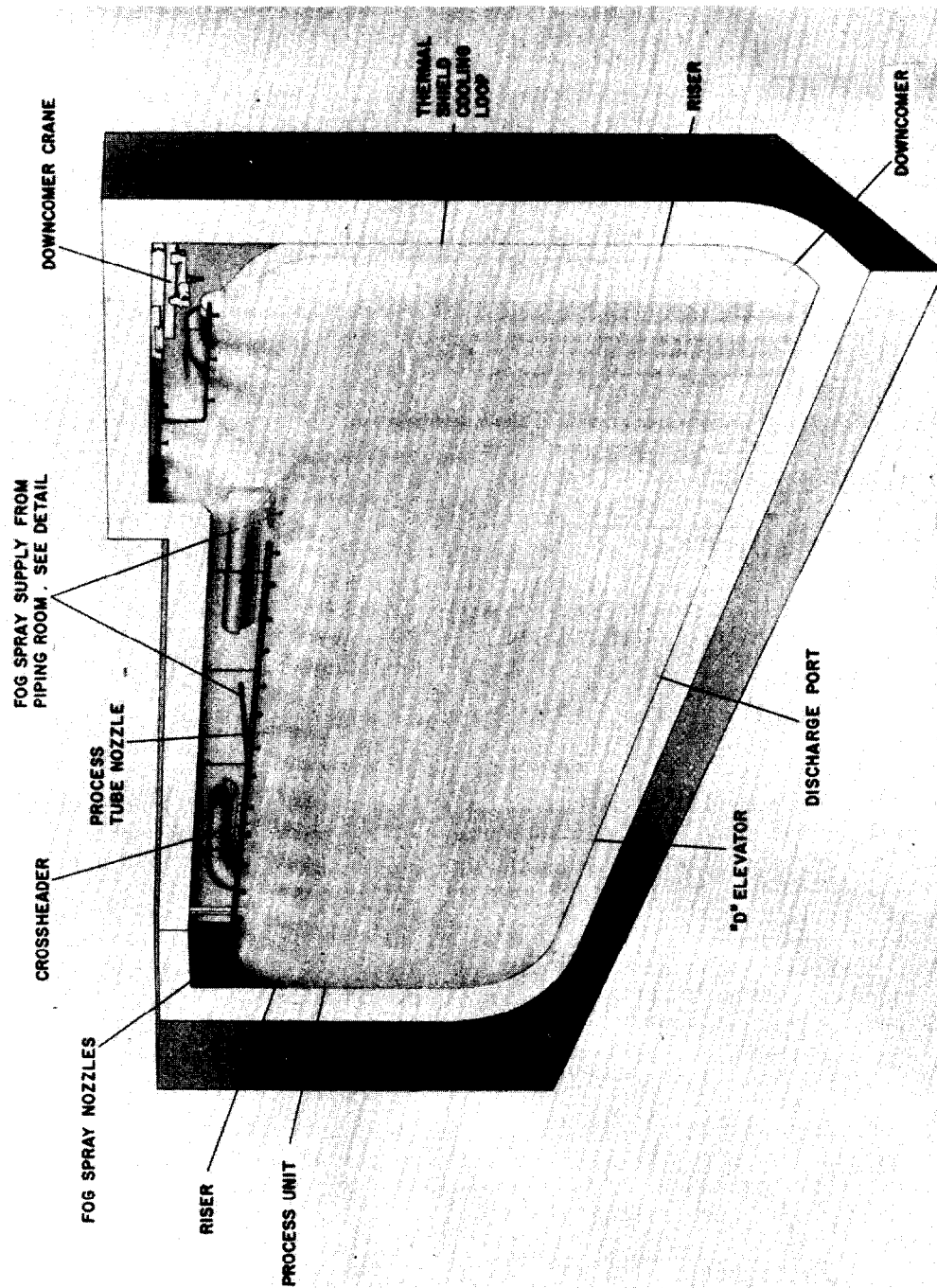


Figure 7 - FOG SPRAY SYSTEM

1. Absorbing a portion of the halogen vapors released during any uranium fires.
2. Settling out a portion of the airborne particulate matter released during fuel element fires.
3. Washing down exposed surfaces within the rear face enclosure for removal of contaminated particulates.
4. Providing some degree of thermal cooling to exposed fuel elements.
5. Condensing any steam that may be formed to prevent unnecessary pressure build-up within this area.

A reliable source of supply water to the sprays was provided by the addition of controls to automatically start back-up steam driven pumps in the event of electrical failure.

This component of the confinement system was the first phase of the work to be completed and was turned over to the reactor plants for use before completion of the rest of the system.

#### Operational Experience

Originally it was planned to activate the fog spray system by a signal from a scintillation detector that measured the concentration of radioactive Iodine in the exhaust air stream. This was proven to be unsatisfactory due to the background fluctuations and secondary radiation of Argon 41. The condition was corrected by the use of a linear differential amplifier that furnished the signal to activate the fog spray.

During the years of operation none of the reactor plants has had a credible incident that would activate the fog sprays when put on automatic operation; however, it has proven to be effective and of value in (1) reducing spread of contamination, (2) reducing air borne contamination during periods of reactor maintenance in the rear face, and (3) cooling exposed irradiated fuel. On these occasions the reactors were shut down and the fog spray system was activated by manual operation from the control room. Utilization of the fog spray system by manual activation during reactor outages has been advantageous as a source of water for rinsing off surfaces of the discharge area after chemical decontamination.

In no instance has the activation of the fog spray system failed when required. This reliability and performance of the equipment at the reactor plants is the effect of mandatory functional checks required by the process equipment standards.

#### System Testing

Each week the Iodine monitoring equipment which provides the trip signal for the fog spray system is functionally tested and calibrated against a known radiation source. The valving to this spray system is visually checked for operability each month, and automatic activation of the valving by tripping the Iodine monitoring equipment is operated at full flow and inspected for evidence of plugging and adequacy of water coverage of the sprays.

By: W. V. Thompson

## THE SRP PRODUCTION REACTOR CONTAINMENT PROGRAM

by

J. A. List

E. I. du Pont de Nemours & Co.  
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### ABSTRACT

A program is under way at the Savannah River Plant to increase the containment facilities as much as practical in the existing production reactor buildings. The major features of the system are outlined, and the air cleaning portion of the installation is discussed in detail.

### INTRODUCTION

Congressional authorization has been given to provide the Savannah River production reactors with additional facilities to minimize the escape of radioactivity in the event of a nuclear incident. This partial containment program will provide the maximum practical protection utilizing the existing building structures.

As I'm sure most of you are aware, the Savannah River reactors are cooled and moderated with heavy water. The reactor buildings are of heavy concrete construction designed to have a certain degree of resistance to an outside blast such as might occur during an enemy attack. The ventilation systems for the process areas are high capacity, once-through systems. The reactors are separated by about 2-1/2 miles and are 5 miles or more from the nearest plant boundary.

In order to decide upon the approach to be used in adding containment to the existing buildings, a great many potential accidents were analyzed. These analyses included not only the situation at the instant of the accident but also considered conditions following the accident up to the time no further operator attention or equipment functioning would be required. The results indicated that most accidents would be self-limiting and that the chance that the pressure capabilities of the building would be exceeded were very remote. On the other hand, many of the less severe accidents could deteriorate into serious situations if not properly handled.

Most of the potential accidents would result in limited fuel melting and a surge of contaminated steam into the Process Room at the time of the accident, possibly followed by additional fuel melting, steam generation, and release of fission products, depending on how well the decay heat was controlled after the incident.

### DESCRIPTION OF THE SAVANNAH RIVER CONTAINMENT SYSTEM

A great many systems were considered during the early Savannah River containment studies. The usual approach of enclosing the reactors in a

containment shell was ruled out by the necessity of using the existing building. On the other hand, it was doubtful whether a filtration system could satisfactorily handle the radioactive steam and fog that could be generated by potential accidents. Decay heat removal also presented a problem, as an analysis of the heat transfer capabilities of the building indicated that very high temperatures would be reached if no means were provided for removing heat from the building. Many possibilities were considered and rejected before it was decided to install the best practical filtration system and, at the same time, to take all reasonable steps to prevent further fuel melting after the initial incident and thus minimize the load on the filters. The system that resulted consists of the following major items:

1. A filtration system to remove a high percentage of particulate matter and halogens from the ventilation exhaust.
2. A system for the control of decay heat to prevent fuel melting after an accident; the system should use the normal cooling system if possible and, if not, provide for direct addition of light water to the heavy water system.
3. Open earth basins to collect contaminated water generated by the addition of light water to the reactor.
4. Remote control facilities to operate certain critical equipment if evacuation of the area should become necessary.
5. Sealing of the Process Room to the extent practical to minimize escape of activity through passages other than the exhaust system.
6. Sprays in the Process Room to cool an irradiated element dropped during discharge from the reactor.
7. Numerous changes to make critical equipment or power supplies more reliable under incident conditions.

Although our interest here is primarily in air cleaning systems, I would like to say a few words about some of the other features before proceeding to a more detailed discussion of the filtration system.

The control of decay heat proved to be a difficult problem. The heat could not be removed by allowing the moderator to boil, as the fuel would quickly become blanketed with steam that could not escape. Recirculation systems handling hot contaminated water did not seem to offer the required degree of reliability.

The direct addition of light water was selected as the simplest and most reliable system; however, the addition of light water at rates sufficient to prevent boiling would generate large quantities of contaminated water that could not be allowed to escape to the Savannah River. Provisions were made to store this water in large open basins, which is definitely not a permanent solution; however, the activity will be retained on the Plant and can be dealt with after the incident is under control.

The remote control system supplies information on conditions in the building and permits operation of certain critical equipment in the building including the light water addition system. The existing system for remote control of the electrical distribution system was expanded for this purpose.

The other features of the system are largely self-explanatory.

## THE AIR CLEANING INSTALLATION

The filtration system consists of high efficiency particulate filters and activated charcoal units. The use of such a combination in a containment system is not new, but the Savannah River system is unique because it is applied in exhaust streams containing mixtures of steam, air, and water droplets. It had been established at Hanford that such a system would handle air containing up to 100% humidity, but it was not certain that the system could survive in an exhaust stream containing steam and entrained water. As soon as the decision was made to proceed with containment at SRP, a program was undertaken to select the best available components for the filtration system and to evaluate its capabilities. The goal set for this program was a system which could:

1. Remove 99+% of the particulate matter and 99.9+% of the iodine from steam or steam and air mixtures;
2. Handle flows per filter unit (normal rating 1000 cfm) of wet steam at 7000 cfm for 30 sec. and 2000 cfm for 5 min. to be followed by a flow of steam and air at 1000 cfm with the amount of steam reduced in steps to follow steam rates characteristic of those produced from reactor decay heat;
3. Have a life of 3 months following an accident; and
4. Provide for remote removal and disposal of filter units.

I'm sure you will agree this is asking quite a lot of a filter system, and we were not at all sure it could be attained. Fortunately, only partial success was necessary to justify the installation. As it turned out we came closer to attaining these goals than was expected when the program was initiated.

Tests with filters quickly established that high quality water-repellent filters would withstand the flows given above as long as they were clean, but when a layer of dirt built up on the filter medium the dirt layer would "blind" with water and the desired flows could not be obtained. Since it was obviously impractical to make the dirt layer water repellent, a high efficiency entrainment separator was added up-stream of the filters. These separators are known as "Demisters" and are manufactured by the Otto H. York Company, Inc. The "Demister" used in the Savannah River system consists of a two-inch-thick mat woven from stainless steel wire wrapped with "Teflon"\* fibers. The "Demisters" are installed at the rate of 5 "Demisters" for 8 filters in order to increase the superficial velocity and obtain a pressure drop of about 0.9 inch of H<sub>2</sub>O across the "Demisters". With this combination, and with an accumulation of dirt on both the filters and "Demisters", flows would drop to 500-600 cfm after the initial steam burst but would recover as the amount of steam decreased. A temporary drop of this magnitude would not have a serious effect on the control of an incident, so these results were considered satisfactory.

Our application of activated charcoal units required high efficiencies in the presence of entrained moisture. The moisture proved to be no problem, as tests with small sealed carbon beds established that a one-inch-thick bed would consistently remove 99.99% of iodine vapor even when saturated with water from the first flow of steam and in the presence of entrained water up to 0.43 lb of H<sub>2</sub>O/filter/min. The problems of obtaining high efficiencies were purely mechanical and consisted of the prevention of bypassing and the elimination of carbon dust which, of course, could carry iodine through the bed.

\* Product of E. I. du Pont de Nemours & Co.

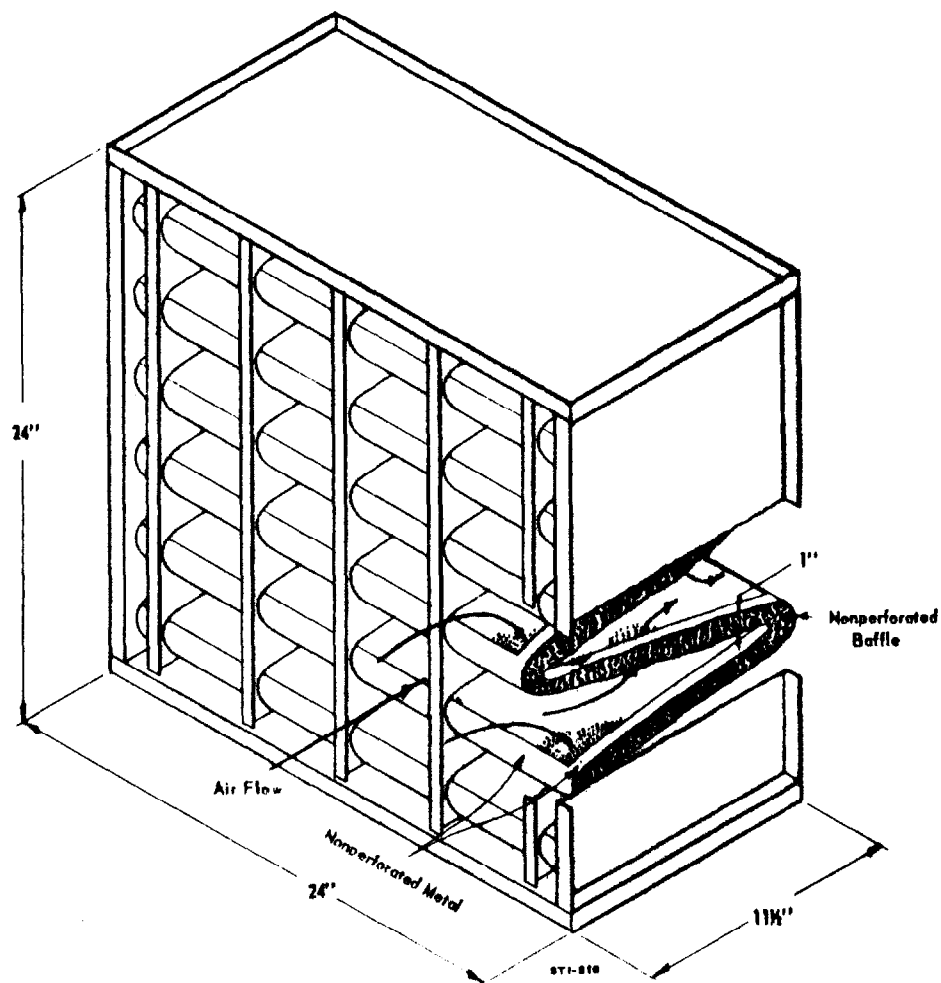
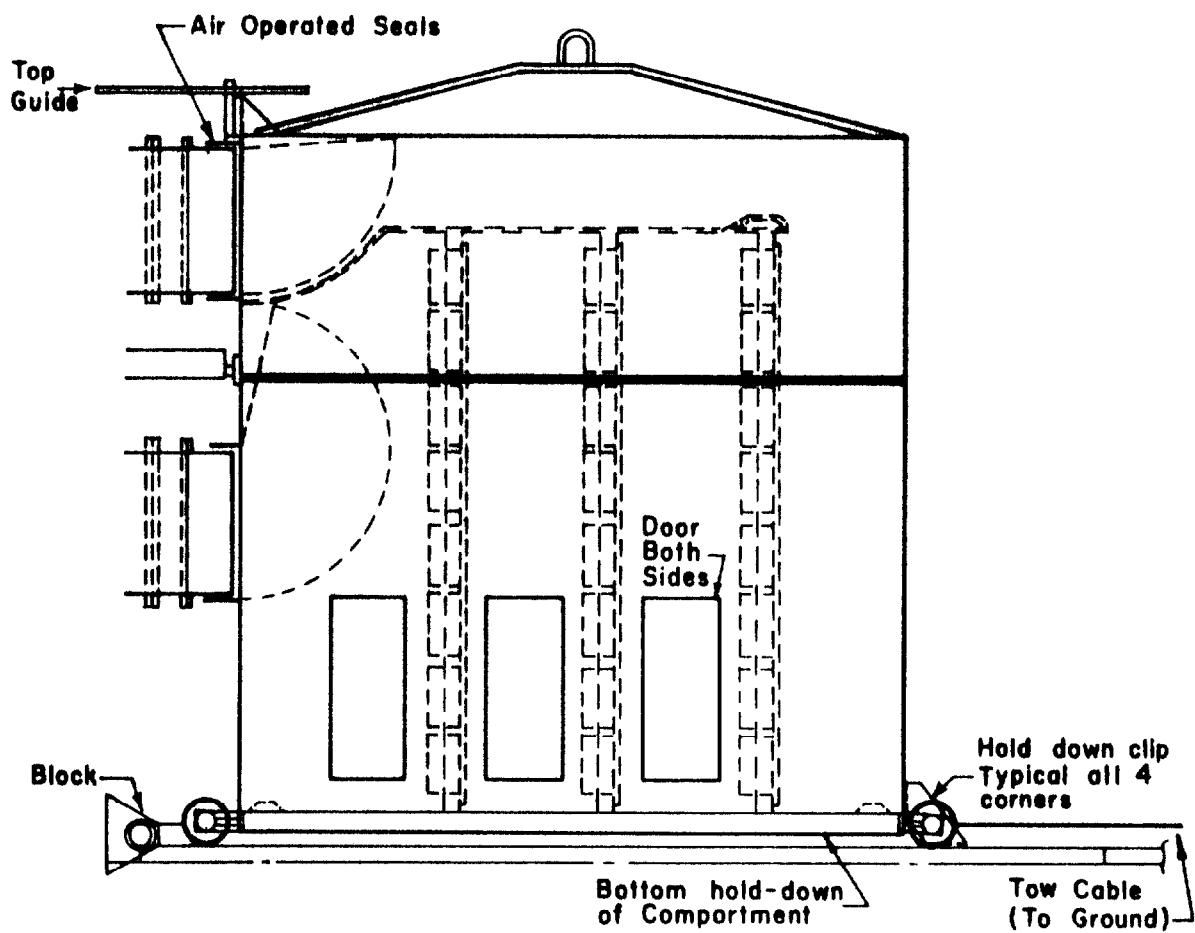


FIG. 1 ISOMETRIC DRAWING OF FULL SIZE CARBON FILTER

Fig. 2  
EXHAUST FILTERING SYSTEM  
FILTER COMPARTMENT



Dusting was minimized by specifying a hard coconut shell carbon, and bypassing was minimized by careful design of the filter unit. Some of the features incorporated in these units are shown by the first slide. You will note that the edges of the screen are unperforated to avoid bypassing at the carbon-to-metal interface; the screen supports are all external for the same reason; the bends are not perforated to avoid high velocity areas; and a support plate is provided at the bends to minimize settling effects. When the carbon units are received, they are subjected to high air flows to detect dusting and to a "Freon" test to detect settling or bypassing. In the first shipment of 186 units no serious dusting was found, and only two units had removal efficiencies below 99.9%. The average of the other units was 99.99%. The two poor units were corrected by adding carbon to eliminate voids.

To avoid excess duct runs, the filter compartments were placed on a roof approximately 50 feet above ground level. Slide 2 shows the system that permits remote removal and disposal. The compartments are placed on dollies and both inlet and outlet are on one end so that the unit can be plugged into the system or unplugged much like plugging a toaster into an electrical receptacle. Contaminated units can be retracted to the edge of the roof, picked up by the shielded crane available on the plant, and hauled by rail to the burial ground. This system should permit disposal of the filters under any conditions permitting recovery of the building. Two other points regarding the compartments are that a space of four feet was allowed between filter banks to allow water drops to settle and to provide access for inspection, testing, and filter change; and that a small trough is provided under the front of each unit to prevent water removed by one unit from running down on the unit below it. To make provision of these troughs easier, the units are attached to the back of the support frame rather than the front as in normal ventilation practice.

A rigid testing and inspection program was followed for both filters and carbon units. A steam flow test was run on a filter fabricated from each lot of filter medium, and every filter was given the usual DOP test after completion. In addition, each installation was given an in-place DOP test, which will be repeated at least annually to ensure continued integrity of the filter system.

Procedures for filling and compaction of the carbon units were rigidly specified, and all units were filled in the presence of a Du Pont inspector. All units were also given an air flow and "Freon" test on arrival at the Plant.

As yet no in-place test for these units has been devised and periodic testing of sample units must be relied upon. However, an in-place "Freon" test is now being developed.

#### SUMMARY

The Savannah River containment filtration system is capable of reducing particulate matter by a factor of 100 and halogens by a factor of 1,000, even in the presence of substantial quantities of entrained water. Under incident conditions, the performance could be expected to be poorer because of such factors as the effect of radiation on gaskets and pneumatic seals, the possibility of overheating the filters or charcoal units, the presence of aerosols, and the possible loss of exhaust fans. However, the combination of the filtration system and provisions for control of decay heat should reduce by several orders of magnitude the probability of an accident affecting off-site areas.

#### DISCUSSION AND COMMENT

Savannah River operated about seven years without containment, and we have gone to it to provide an extra degree of protection since we are operating to closer limits. The cost has been in the range of eighteen million dollars for five reactors. The decision to go over was primarily an AEC one.

The de-misters are woven from stainless steel wire that are wrapped with teflon fibres; the weave is actually a combination of the two.

Session Chairman: R. D. Modrow, of Phillips Petroleum Company, has agreed to discuss the paper, "Removal of Particulates from the Waste Calcination Facility Off-Gas," by L. T. Lakey, J. A. Buckham and Mr. Modrow.

# REMOVAL OF PARTICULATES FROM WASTE CALCINING FACILITY OFF-GAS

by

R. D. Modrow  
L. T. Lakey  
J. A. Buckham

## SUMMARY

This paper discusses the removal of particulate matter from off-gas from the Waste Calcining Facility (WCF), a unit that will convert radioactive liquid wastes to granular solids in a heated, fluidized bed. Descriptions of the off-gas cleaning equipment employed are given together with data for each unit. Also discussed are sampling of the off-gas for particulate matter and other special techniques used in evaluating the system performance.

The overall decontamination factor for the off-gas cleanup system, consisting of a cyclone, a spray tower, a venturi scrubber and separator, silica gel beds, and AEC filters, was about  $3.6 \times 10^5$ . This resulted in a reduction of the solids loading from approximately 1.4 grains/acf to about  $5 \times 10^{-6}$  grains/acf; particulates in the final off-gas were essentially spherical and were less than 0.1 micron in diameter. This high degree of decontamination indicates that a noncondensing-type cleanup system can be used to reduce radioactive particulate contamination in a gas stream to a level low enough to permit the gas to be released safely to the atmosphere.

The use of a radioactive tracer in the system indicated the occurrence of a delayed-type of penetration of the AEC filters by the submicron particles. About ten days were required to reach an equilibrium value of penetration through the filters.

## INTRODUCTION

The Waste Calcining Facility (WCF) utilizes the fluidized bed calcination process for the conversion of high-level aqueous radioactive waste solutions, resulting from reprocessing of aluminum-uranium alloy fuels, into compact granular solids. Treatment of the off-gas from the calciner to remove entrained fine solids is effected in a noncondensing-type system which does not result in the formation of additional liquid wastes. Initial development<sup>(1)</sup> of this continuous process by the Atomic Energy Division of Phillips Petroleum Company began in 1955 and led to the design<sup>(2)</sup> and construction of the WCF at the National Reactor Testing Station (NRTS) in the period from 1958 to 1960. Design and operating data in support of the WCF have been obtained from 6- and 12-inch diameter and two-foot-square pilot plant calciner units<sup>(3,4,5,7)</sup>. Early in 1961, cold testing of the WCF began with the use of a non-radioactive simulated aluminum nitrate waste solution, and was conducted intermittently over a total of 3700 hours of operating time. Late in 1962, cold testing was concluded with a continuous

month-long operating period during which a radioactive tracer was added to the feed to facilitate the quantitative determination of the effectiveness of the off-gas cleaning system.

In the fluidized bed calcination process, as exemplified by the WCF, waste solution is pneumatically sprayed at a nominal rate of 80 gph into a four-foot diameter by six-foot deep fluidized bed of alumina maintained at 400°C. An inlet fluidizing velocity, based on the superficial cross sectional area of the calciner vessel, of 1.0 ft/sec is generally used and a freeboard of about six feet, in addition to a louvered baffle, is provided for de-entrainment of solids from the off-gas. The bed level is maintained by adjusting the rate of withdrawal of product. The calcination of the waste solution to granular alumina is accompanied by the release of large amounts of water vapor and gaseous products. These vapors and gases, along with air employed for fluidizing, feed atomizing, and purging, sweep a portion of the bed material, mainly the fine, irregularly-shaped particles, into the off-gas piping. The initial separation of these solids from the gas takes place in a cyclone which is followed by a wet scrubbing system, which includes a spray tower together with a venturi scrubber and its associated entrainment separator. In the scrubbing system, sufficient condensing is done to allow a scrubbing solution recycle rate of 10-20 gph back to the feed tanks. For a final treatment following the addition of superheat, the off-gas is passed successively through four silica gel beds in parallel and then through three AEC filters, also arranged in parallel. A schematic flowsheet of the WCF is shown in Figure 1.

Treatment to remove nearly 100 percent of the radioactivity from large volumes of off-gas is necessary in the WCF before the off-gas is released to the atmosphere. The calcination of Idaho Chemical Processing Plant (ICPP) aluminum nitrate wastes at 400°C results in two forms of radioactivity in the off-gas: vaporized ruthenium tetroxide, which is largely adsorbed in the silica gel beds, and fine solid material, which has essentially the same radioactive nuclide distribution as the bed material. In order to substantiate ruthenium decontamination factors that have been determined in previous pilot plant tests<sup>(6,7)</sup>, the removal of ruthenium tetroxide by the off-gas system will be determined during initial radioactive operation of the facility. This report will discuss in detail only the removal of particulate matter from the off-gas as determined by using a radioactive tracer during operation of the WCF with simulated feed.

#### DATA COLLECTION

The off-gas was sampled for solids at various points in the system by passing a measured sidestream of the off-gas through a Millipore HA filter, a membrane-type filter with a dust removal efficiency of > 99 percent (8,9) for submicron particles. The sample probes were made from one-half-inch diameter stainless steel tubing with a one-inch taper to a sharp edge. The sample lines, which were from 10 to 35 feet in length, as well as the Millipore filter holders, were heated to preclude condensation. A typical sampling setup is shown schematically in Figure 2.

Four sampling stations were used to measure the rate of solids flow. They are located downstream from the venturi scrubber, downstream from the silica gel beds, and at two places downstream from the AEC filters. Equipment vent gas, which is mainly air used to sparge the contents of vessels not directly connected to the process system, enters the off-gas system between the last two sampling stations. The locations of these stations are shown schematically in Figure 3. The gas velocities at the face of the sample filters were approximately 10 cm/sec, except for one collection station downstream of the AEC filters which operated at a face velocity of 150 cm/sec. The off-gas sampling was nearly isokinetic, even though the particles were sufficiently small (< 0.3 micron) that it was not

# FLUIDIZED BED WASTE CALCINING FACILITY

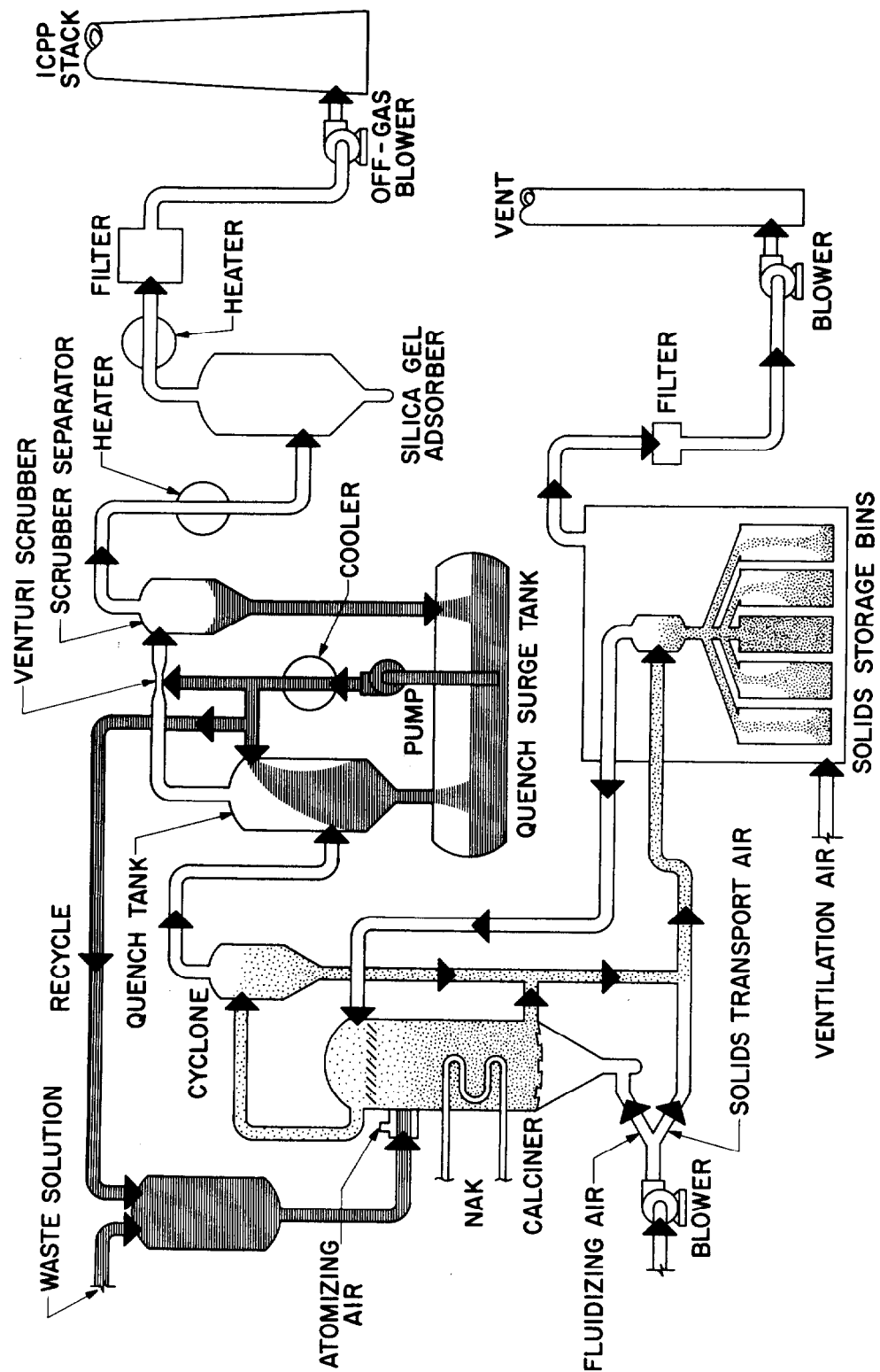
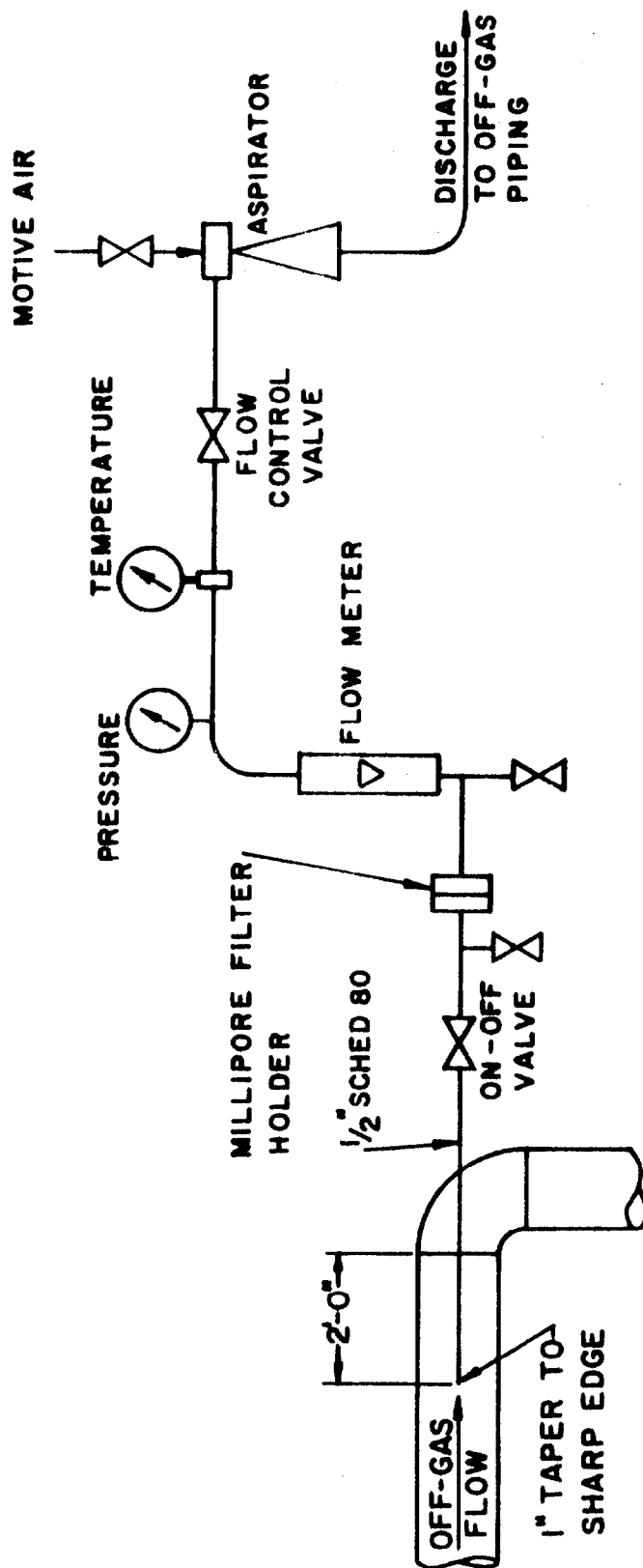


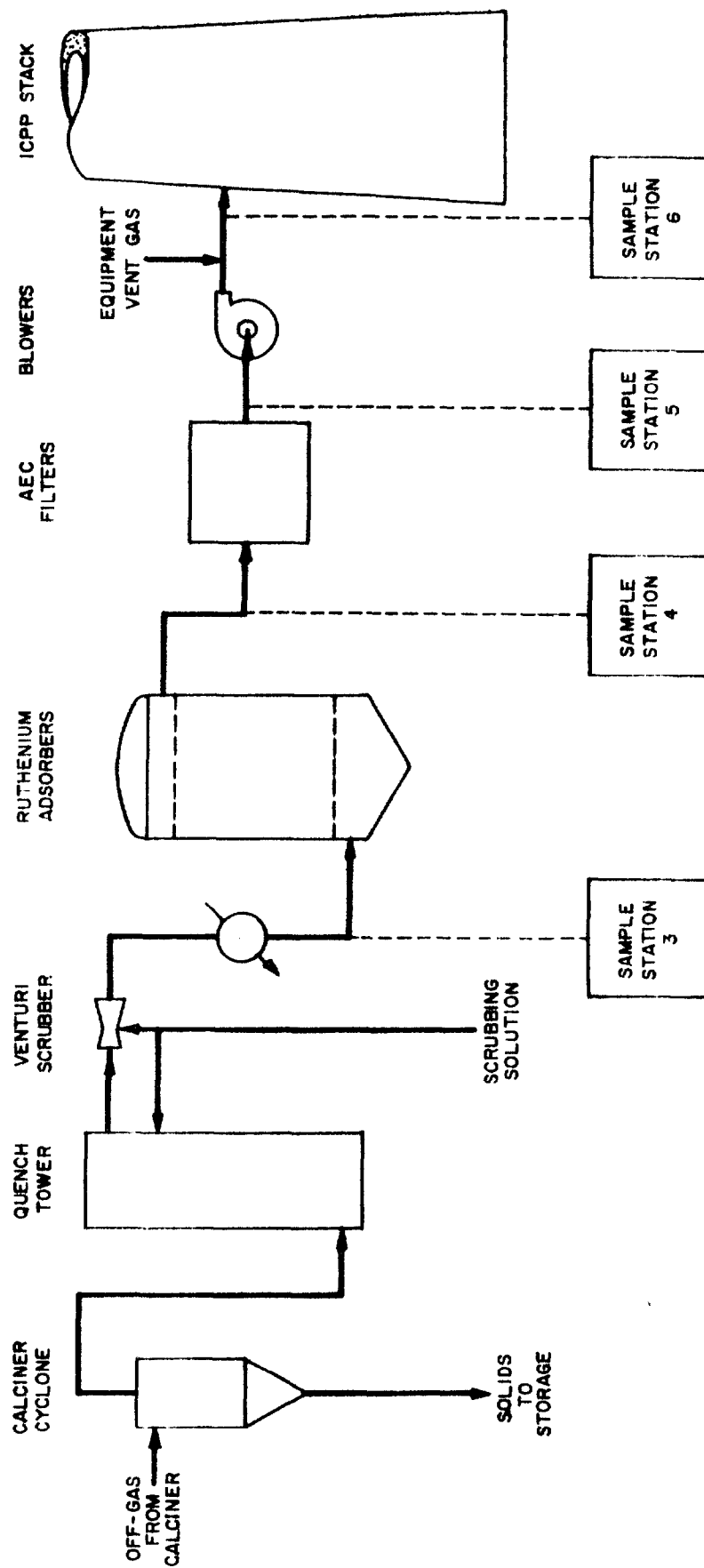
Figure 1. Fluidized Bed Waste Calcining Facility



NOTES: ALL LINES AND FILTER HOLDER ARE HEATED  
ALL MATERIAL STAINLESS STEEL

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Figure 2. Typical Off-Gas Sampling Setup



CPP-S-2731

Figure 3. Location of Solids Sampling Stations

necessary<sup>(10)</sup>. In all cases the stream velocities were greater than the sampling velocities, so that larger particles, if there had been any, would have had a tendency to be carried into the probe in disproportionately high amounts.

After the solids were collected for a predetermined time, varying from 4 to 24 hours, depending on the solids concentration at the particular point, the filters were removed from the filter holders and the amounts of solids were determined by at least one of three different methods. One method, gravimetric, was to ash the filter plus the plastic bag used to transport the filter, and weigh the residue. This technique worked satisfactorily for samples collected only at sample station 3 which is downstream from the venturi scrubber, since the weight of solids collected at other sample stations was too low for such measurements to have practical significance. Another method, tracer activity, was used to correlate the activity of solids collected downstream from station 3 with the solids weight-to-activity ratio at station 3, where both gravimetric and tracer activity measurements were made. Because the sodium-24 tracer activity decayed beyond the detection limits within four days at the downstream sample stations, a third method, activation, was used in which the sodium normally in the solids was activated by irradiating the filters together with a standard alumina sample in the MTR, a nuclear reactor. The decay disintegrations of the solids on the filters were then counted, as in the tracer activity method, and compared with the activated standard alumina sample to arrive at a sample weight.

Spot checks of the sizes of particles collected at the four sampling stations in the WCF off-gas system were made with an electron microscope. The particles were all submicron in size, ranging from 0.3 micron in diameter to less than 0.01 micron, and all appeared to be essentially spherical in shape. Preparation of the sample for viewing involved the transfer of a small amount of partially ashed filter to a pollodion-covered electron microscope grid. Identification of the particle shape as well as estimation of the sizes of the particles was enhanced through the use of stereophotographs of the images. The size distributions of the particles found in the off-gas system are given in Table I.

## PERFORMANCE

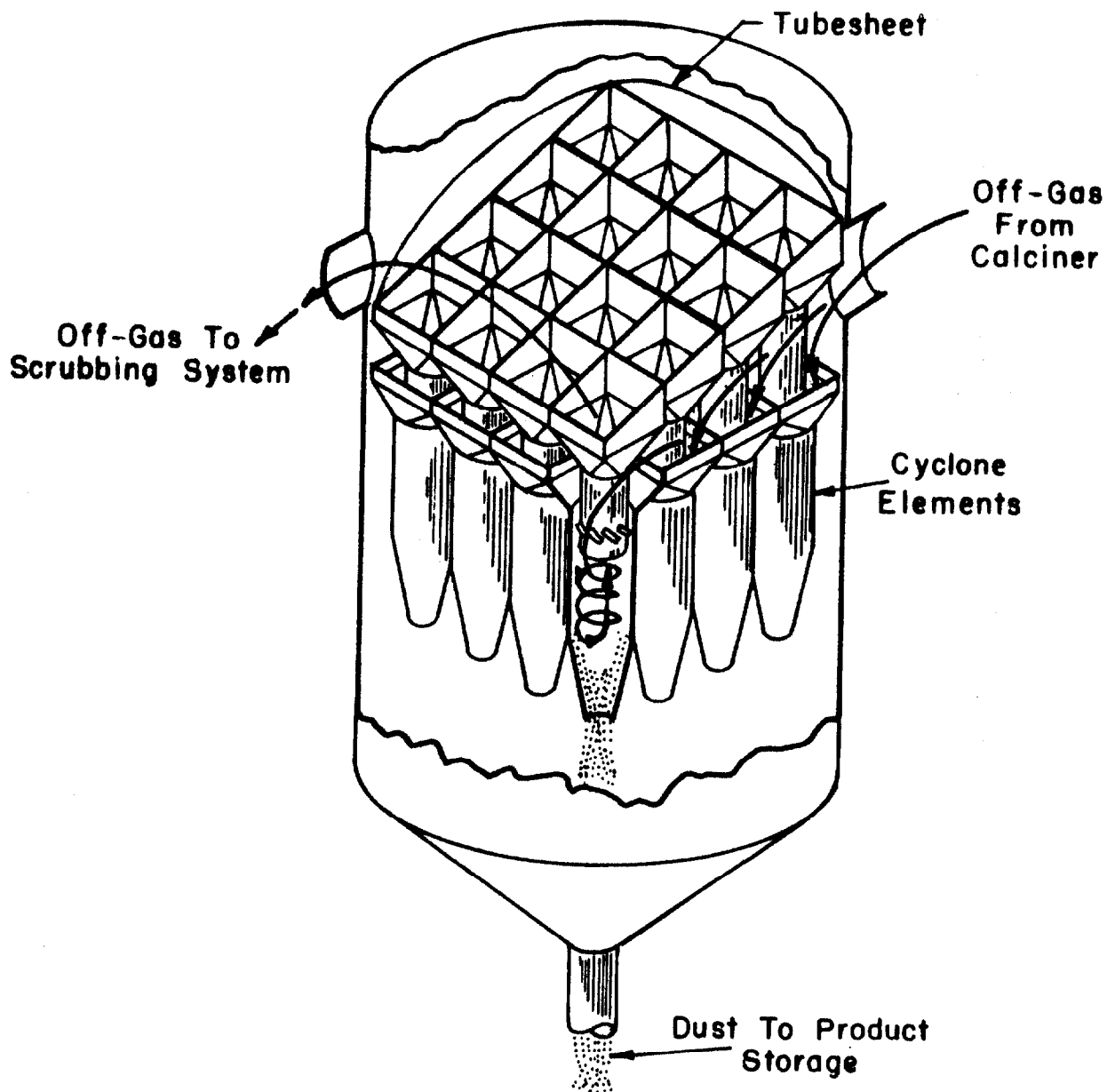
Removal of Dry Fines: Initial removal of the solids from the calciner effluent gas is performed in a Multiclone, a parallel arrangement of small cyclones with a single inlet and a common fines hopper. The system generally has been operated so as to give a cyclone decontamination factor (DF) of 4, which is equivalent to 75 percent efficiency and is the ratio of the inlet to the outlet flow of solids in the off-gas.

Although the original Multiclone unit had 16 elements, only 6 to 9 of these 6.5-inch diameter cyclone elements have normally been used and the rest have been blocked off because of a change in operating requirements. The original cyclone is shown in Figure 4. The off-gas inlet velocity to the individual elements has generally been 50 ft/sec when using six of the elements and the cyclone DF has ranged from 3.2 to 13.5, depending mainly on the solids loading in the inlet gas. At a nominal inlet flow rate of 29 lb/hr of solids, ranging in effective diameter from 2 weight percent larger than 700 microns to 10 weight percent smaller than 10 microns, the cyclone DF has been about four.

Fines collected in the calciner cyclone are currently sent to solids storage along with the bed product to give a combined product that is 97 weight percent larger than 100 microns in effective diameter. In early studies, the fines collected in the cyclone were returned to the bed, but this resulted in an undesirable four-fold increase in the rate of flow of solids to the cyclone and up to a two-fold increase in the rate of carryover of fines past the cyclone to the scrubbing system.

TABLE I  
SIZE DISTRIBUTION OF PARTICULATE MATTER  
COLLECTED DOWNSTREAM OF THE WCF VENTURI SCRUBBER  
ON MEMBRANE-TYPE FILTERS

<u>Location</u>	<u>Sampling Station Number</u>	<u>Number of Particles Measured</u>	<u>Average Particle Diameter (Microns)</u>	<u>Range of Particle Diameters Observed (Microns)</u>
Downstream of Scrubbing System	3	47	0.10	0.01 to 0.26
Downstream of Ruthenium Adsorbers	4	27	0.04	0.02 to 0.20
Downstream of AEC Filters	5	49	0.04	0.006 to 0.13
Downstream of Equipment Vent System Inlet to The Process Off-Gas System	6	22	0.045	0.01 to 0.11



CPP-S-2839

Figure 4. Calciner Cyclone Prior to Modifications

Removal of Wet Fines: Secondary removal of solids from the calciner effluent gas occurs in the scrubbing system where an average DF of 900 was attained during the latest extended test run of about one month duration. The scrubbing system consists of a spray tower, a venturi scrubber with its associated separator, and a heat exchanger. The complete system is generally operated at a pressure drop of approximately 70 inches of water. Virtually all of the 7.2 lb/hr of the incoming solids, 90 weight percent of which are smaller than 75 microns in effective diameter, are removed by the scrubbing system; the flow of solids downstream is reduced to only 0.008 lb/hr of particles, all of which are smaller than 0.3 micron in diameter.

The scrubbing solution, which at equilibrium conditions is about 5M nitric acid, 0.8M with respect to aluminum, and contains a small amount of undissolved alpha alumina, is pumped to both the spray tower and the venturi scrubber. About 63 gpm of cooled solution is sprayed into the spray tower to reduce the temperature of the off-gas before it enters the venturi scrubber. The 15 gpm (13 gal/1000 acf of off-gas) of cooled scrubbing solution injected into the venturi scrubber through four 0.25-inch diameter holes intimately contacts the suspended particles in the off-gas which has a velocity of about 300 ft/sec and causes some of the nitric oxides and water vapor to be condensed. Separation of the particle-laden scrubbing solution from the off-gas occurs in the entrainment separator which is followed by a heater to prevent further condensation. The condensed scrubbing solution is recycled to the feed at a rate necessary to prevent an excessive buildup of either dissolved or undissolved solids in the solution. The venturi scrubber and the entrainment separator are shown in Figure 5.

After passing through the cyclone and venturi scrubber with its associated separator, the effluent gas has been freed from the bulk of the solid material it contained when it left the calciner. Solids removed up to this point have been present in sufficient quantities to require continuous disposal or recycle. However, the quantity of solids remaining in the off-gas downstream from the venturi scrubber-separator is very small and the problem from this point on is primarily one of cleanup of the off-gas stream for disposal.

Final Off-Gas Cleanup: The concentration of particulate matter in the off-gas leaving the venturi scrubber-separator is reduced a total of 100-fold in passing through the silica gel beds and the AEC filters. The superficial off-gas velocity through the silica gel beds, across which a particulate DF of about 5 was obtained, was approximately 0.2 ft/sec, based on the area of three of the four units operated in parallel. The AEC filters operated at a superficial face velocity of about 2 ft/sec (a velocity of about 0.04 ft/sec through the filter fibers) with an average inlet particle diameter of about 0.04 micron. Initial indications were that the AEC filter DF was greater than 1000 just after introduction of the radioactive tracer; however, at equilibrium the DF had decreased to about 20.

The silica gel beds were installed primarily for the adsorption of ruthenium tetroxide which is volatilized initially from the calciner<sup>(6)</sup>; even so, they had a DF of 5 for particles with an average diameter of 0.1 micron. At least two-thirds of the ruthenium vaporized from the calciner is expected to be removed from the off-gas with the fines collected in the cyclone and with the scrubbing solution<sup>(7)</sup>. Ruthenium tetroxide vapor passing downstream from the scrubbing system will be largely adsorbed, with an expected DF of at least 1000, in the silica gel beds<sup>(6,7,11)</sup> which are seven-foot diameter by three-foot deep beds of 6- to 12-mesh particles. Ruthenium tetroxide vapor that condenses to particulates is expected to be filtered out either by the silica gel or the AEC filters.

The DF of the AEC filters was about 20 even though the particulate matter had an average diameter of only about 0.04 micron. Each of the three parallel units consists of a prefilter, specified to have a DF of 1.5 (35 percent efficiency)

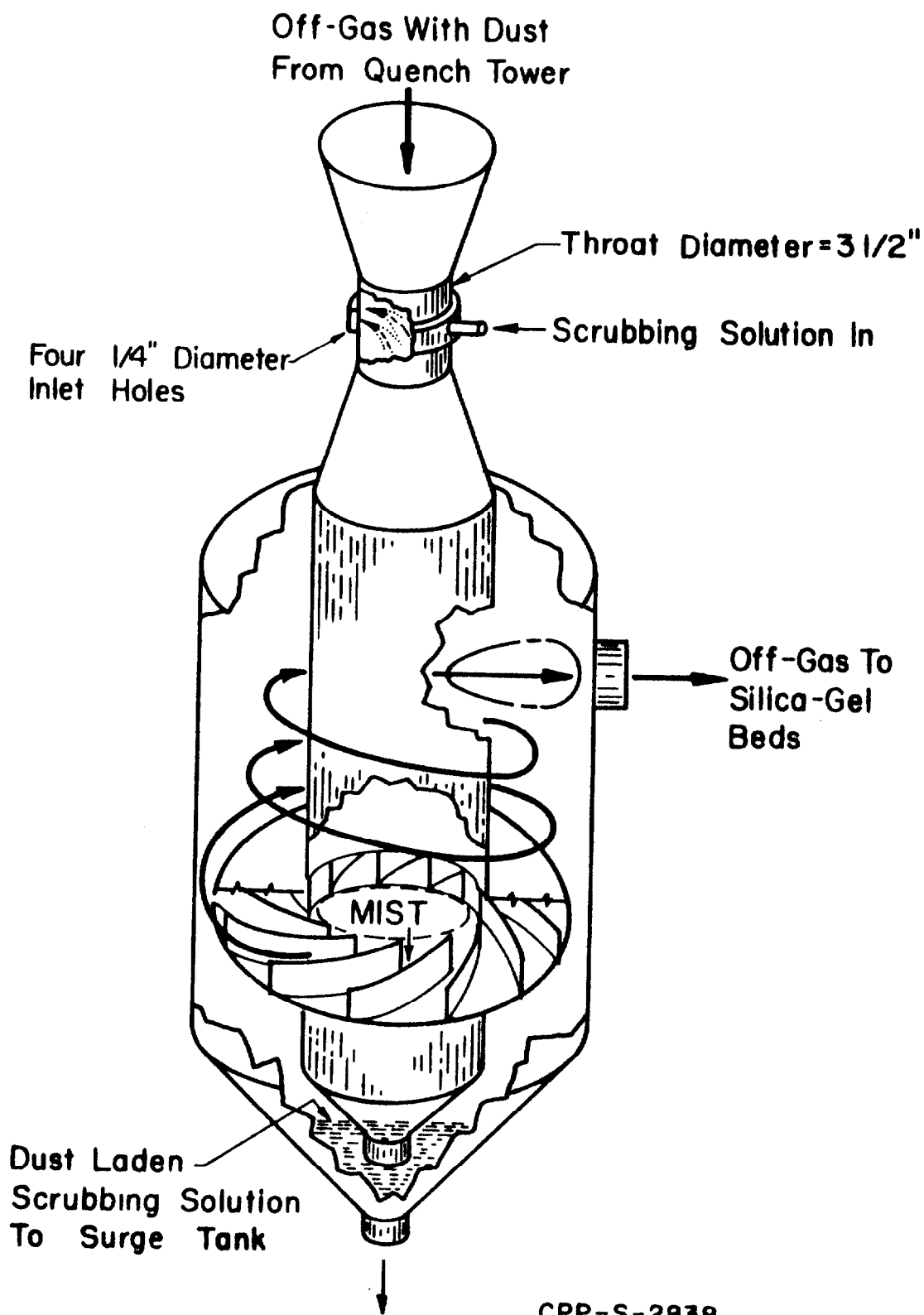


Figure 5. Venturi Scrubber and Entrainment Separator

by the NBS discoloration test, and a final filter, specified to have a DF of 3333 (99.97 percent efficiency) by the DOP mist test<sup>(12)</sup>. Just prior to the 30-day test run, the AEC filter installation was tested with 0.9-micron DOP mist particles and was found to have a DF of 10,000 for this material.

Shortly after the addition of the tracer to the system, a DF of over 1000 was indicated from measurements of solids in the off-gas taken upstream and downstream from the filters. Over a period of about 10 days, as shown in Figure 6, the indicated DF decreased before stabilizing near a value of 20. It would appear that these filters block the passage of a large number of the particles in the entering gas stream, but serve only as an impedance to the passage of other particles. The filters were in use for 14 days immediately prior to the time that the data were collected, so an equilibrium holdup of very small particulate matter that did not contain any radioactive tracer had probably already been established in the filters. The data from the two sample stations upstream of the filters did not exhibit the delayed penetration characterized by the data from the downstream sample stations; rates of solids carryover past both the scrubber and the silica gel beds rose almost instantaneously to the equilibrium level as exemplified in Figure 7 for carryover past the silica gel beds.

Additional evidence of the delayed penetration of the filters was obtained from post-run samples; the calciner was shut down and the off-gas piping was cut to allow fresh air to be pulled through only the silica gel beds and the AEC filters while sampling at stations 3, 4, and 5. During this week-long test, the solids loading downstream of the filters remained fairly constant at the indicated steady-state value (see Figure 6). During the same period, however, the solids loading of the air entering the filters dropped rapidly (see Figure 7); thus the re-entrainment of particles from the silica gel beds was negligible.

## CONCLUSIONS

Removal of particulate solids from the WCF off-gas was very good, with a DF of  $3.6 \times 10^5$  measured from the outlet of the calciner to the stack. This DF, in addition to a conservative atmospheric diffusion factor<sup>(14)</sup> of  $10^{-4}$  sec/meter<sup>3</sup>, is expected to keep the concentration of strontium-90 to about 1/13 of the Maximum Permissible Concentration as set forth in NBS Handbook 69<sup>(13)</sup>. The performance of the venturi scrubbing system, which alone had a DF of 900, was excellent. Individual DF's are shown schematically along with the particulate flow rates in Figure 8.

A delayed penetration of the filters was indicated from tracer activity data collected during a 30-day operating period. This finding indicates the necessity for observations of filter performance in a test of this type over an extended period of time before drawing conclusions regarding the efficiency of such filters.

The particle size of solids escaping collection by the venturi scrubber system averaged 0.1 micron in diameter as determined by viewing the particulate matter collected on membrane-type filters with an electron microscope. After passage through the silica gel beds and AEC filters, the average particle size was reduced to about 0.04 micron.

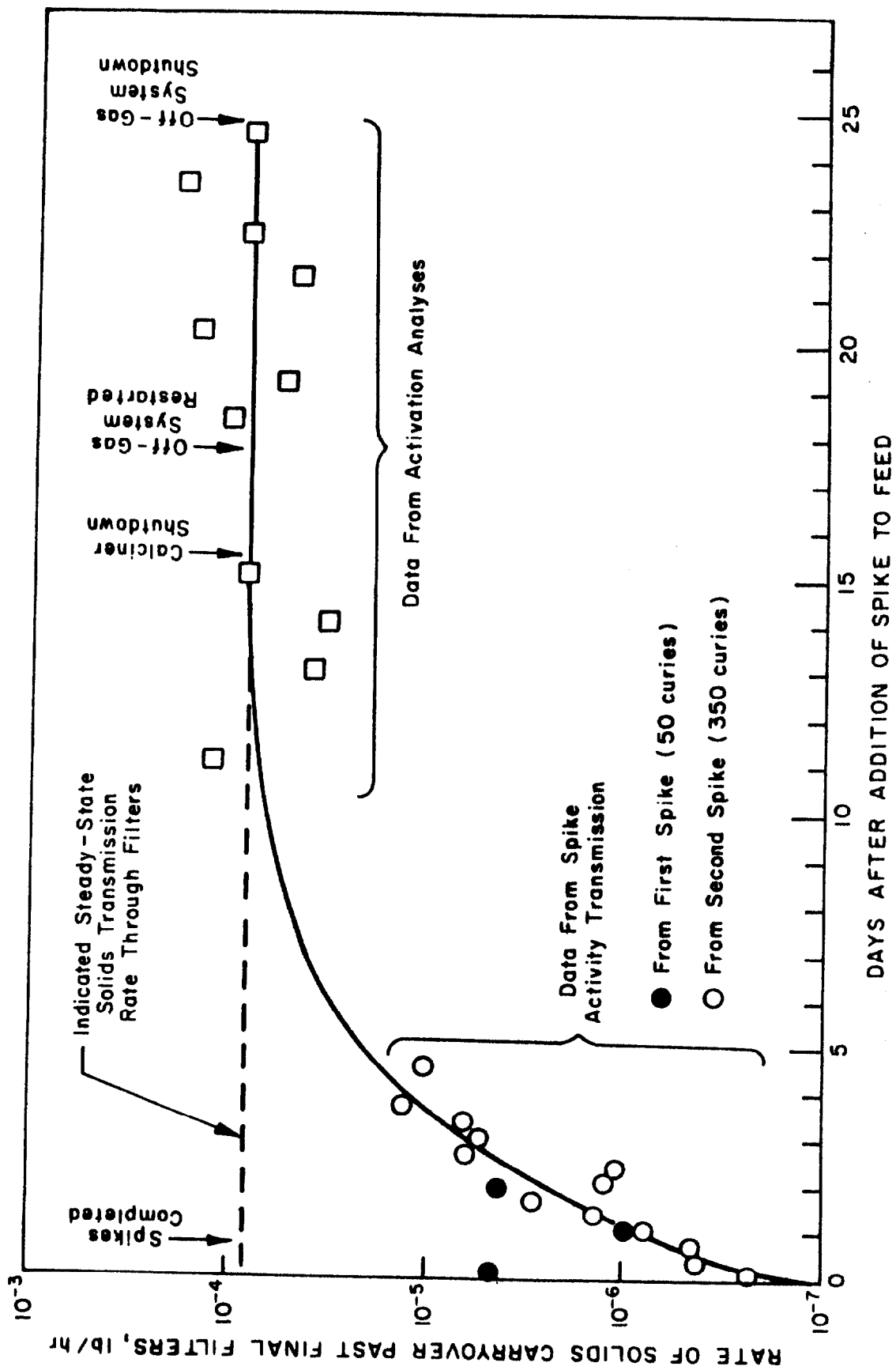
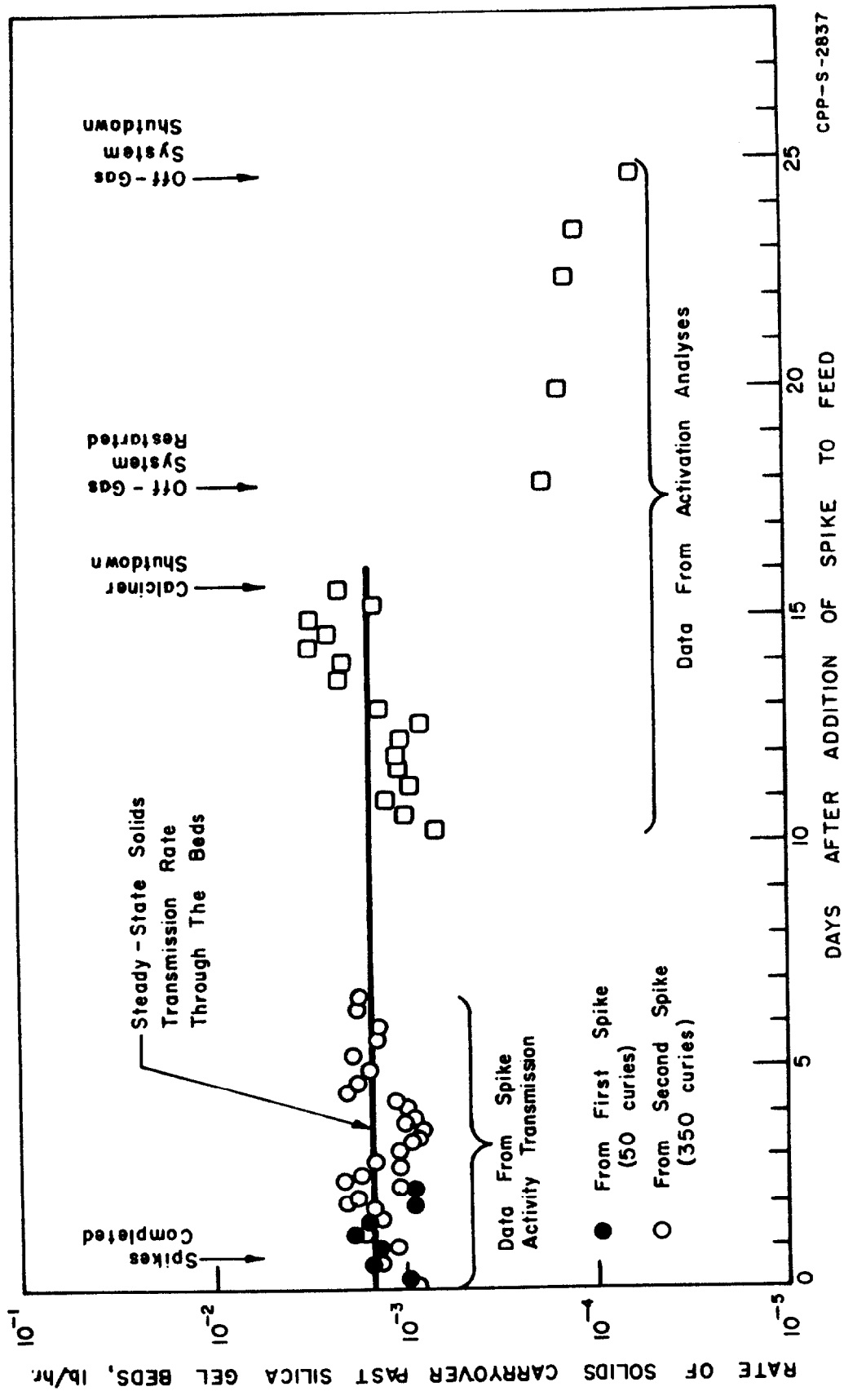


Figure 6. Solids Flow Rate Past the AEC Filters

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Figure 7. Solids Flow Rate Past the Silica Gel Beds

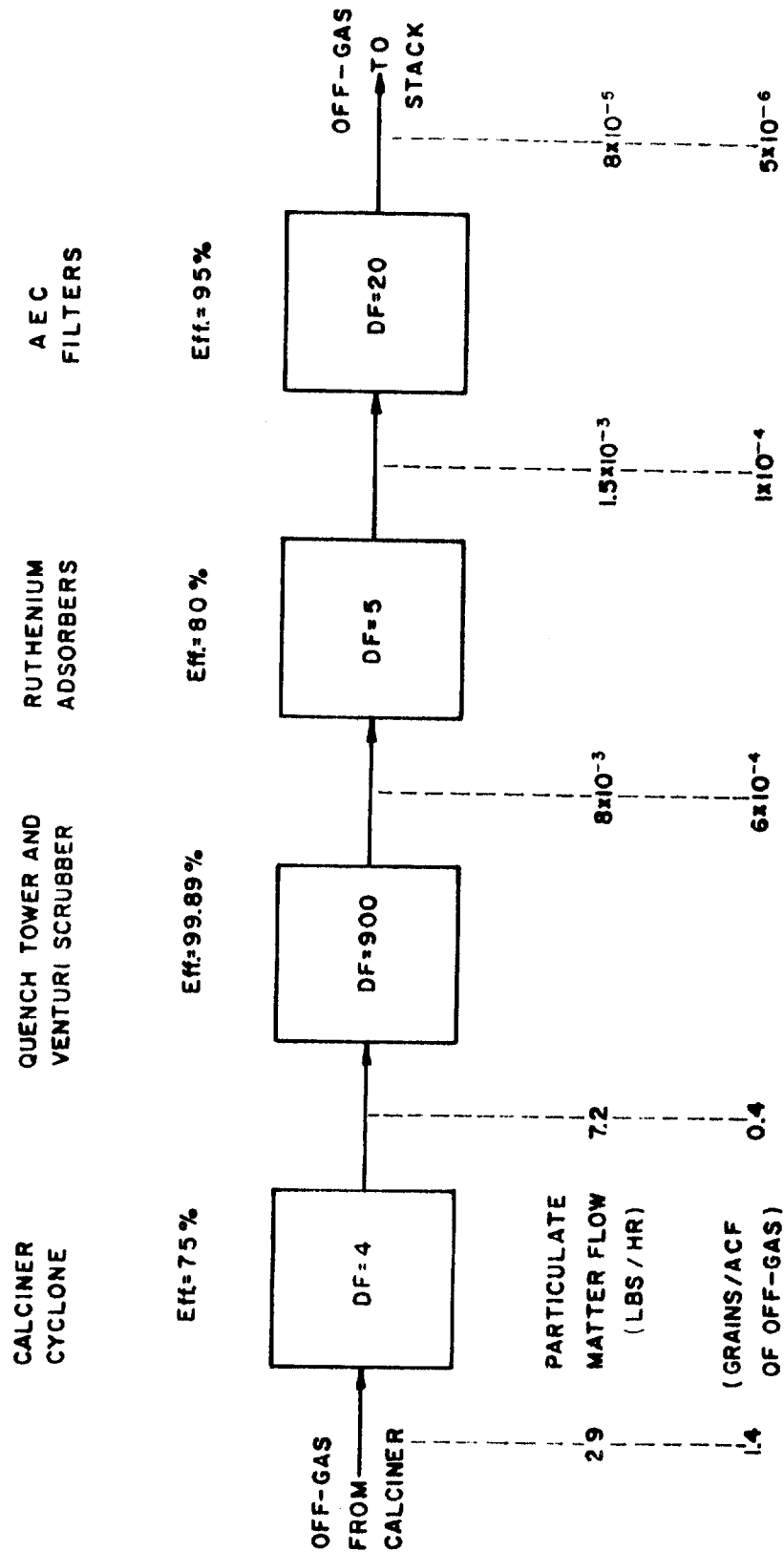


Figure 8. Decontamination Factors (DF) and Particulate Flow Rates for the Off-Gas System

CPP-S-2730

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## DISCUSSION AND COMMENT

HACL Comment: I am a little perplexed by the results on the absolute filters, because I think all the data we have indicate that as they load up, their performance gets better, and I think Mr. Modrow's supervisor wrote me about this, a question of whether or not this phenomena could be attributed to anything else. One possibility is mechanical vibration and migration of the particulates. But at the loadings that are loaded in the particle sizes, I think are less than the DOP sizes. What we would be concerned with is whether or not the adhesive forces in this system are some way reduced, and therefore particles do migrate. But I still am perplexed as to why, with continued use, the loading and deposition which we have always seen happen in every filter in the business. In fact, all AEC filters that have been on-stream and have been tested after they have been plugged a little seem to produce DF's that are almost of the order of one magnitude better because of the deposition. So this, I think, may be in the realm of speculation, but I would like to hear more about it.

Comment by the Speaker: I don't have any direct answer to that. The only thing that comes to my mind is that we are working here with a real dry particulate matter, rather than atmospheric dust, which most of the tests have been made with, or with a mist such as DOP. This could have something to do with it.

The filtering velocity, of course, is fairly low. The filtering velocity is about .04 of a foot per second and a face velocity of 2 feet per second, which is lower than normal. This could have something to do with it, too.

I was wondering, myself, if anyone else has seen this type of phenomena with the filters. I would be interested, too, in finding out exactly what the phenomena is, here, in this delayed penetration.

These filters were in use for two weeks prior to the addition of the radioactive tracer to the feed. The pressure drop over the total filtering system, which includes a box which holds both the prefilter and the absolute filter, registered about one inch of water, and some of this, I am sure, was due to the inlet and outlet valves in the system, and no pressure drop increase could be noted across the filters during this 30-day operating period.

HACL Comment: I have two points, I think, on which an answer can be postulated. One is that the lowered velocity here for the size range you are using ought to favor removal efficiency, rather than hinder it, because you are in the diffusion range.

Secondly, the other thing that occurred to me, and I think I wrote Dr. Hoffman about it, is the question of pulsation; possible pulsation flow induced in the system by the fan; if you have a little bit of vibration into the system, you might in fact get some agitation of the bed. These filters aren't as rigid as you might think, and could cause the migration. At any rate, I still would like to know if in your test setup you are going to follow this for more than the ten days? I think that the curve will start down again.

Comment by the Speaker: Could I answer the last question first? We are going to try to determine if there is any pressure fluctuation in the system in the forthcoming tests. We have centrifugal blowers just ahead of the stack, as you probably saw in the flow sheet. We don't see how this could affect the filters which are fairly close to the centrifugal blowers.

Do you think that a centrifugal blower would cause enough pulsation?

HACL Comment: As far as I know, we have some pretty good evidence in certain bag filters with centrifugal blowers, that if you get the blower and distance just right to the bag collector that the bags are vibrating all the time, the dust keeps coming off of them, and they tend to self-clean. So, I would believe that you could get air-column vibrations that would appear certainly at the surface, the back surface of the filter. This is really not steady flow, if you analyze it. It has got the cycle chopping of the centrifugal vane.

Comment by the Speaker: The filters were tested in place, just prior to this 30-day run with the DOP mist test. The DOP test size in this case was about 0.9 microns; and the efficiency of the filters with this test showed about 99.99+. A DF of about .020.

Session Chairman: I should like to thank Messrs. Anderson, Arnett, Green, List and Modrow for a very stimulating number of papers, as evidenced by the questions and discussions.

## SESSION VI - SPECIAL PROBLEMS: RARE GASES AND INCINERATORS

Morning - 23 October 1963

A. J. Breslin, HASL, Chairman

Session Chairman: This is Session VI - Special Problems: Rare Gases and Incinerators. The ground rules are as before. I will ask you to hold your questions and discussion period until after the three papers have been presented.

The first paper is entitled, "Use of Foam for Containment and Cleanup of Radioactive Aerosol and Gases," by R. E. Yoder and L. Silverman, HACL, and will be presented by Dr. Silverman.